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UK ABWR Generic Design Assessment

Generic PCSR Chapter 25 : Probabilistic Safety Assessment

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Executive Summary

This analysis chapter summarises the Probabilistic Safety Assessment (PSA) of the UK ABWR that has been performed for GDA. It provides an integrated and structured safety analysis that combines consideration of engineering design and operational features in a consistent framework for the assessment of plant risks.

A full scope PSA model has been developed that considers all sources of radioactivity at the facility; all types of initiating events that could lead to radioactive release (internal events, internal hazards and external hazards), and all operational modes. Together, these elements define the magnitude of the radiological hazards and resultant risks.

The PSA is comprised of 3 levels. Level 1 PSA focuses on the potential for core damage or for non-reactor fault plant damage states. Level 2 PSA widens this analysis to consider release magnitudes and frequencies from loss of the containment function. Finally, Level 3 PSA is wider still, and considers risks to the public from offsite releases.

This chapter provides an overview of the PSA model development and the analysis approach, methodology including use of computer codes, and documentation. It presents a summary of the results of quantification of risks and an evaluation of insights that have been used in assessments to demonstrate that risks are As Low As Reasonably Practicable (ALARP).

The overall risks calculated for the facility are compared against appropriate Risk Targets for the UK, as defined in the Nuclear Safety and Environmental Design Principles. The chapter also considers other risk metrics used internationally, such as Core Damage Frequency and Large Release Frequency. The PSA results are used to assess and document the strengths and weaknesses of aspects of the design with complex systems and interdependencies, and to support the evaluations of potential plant risk reduction measures as part of the ALARP demonstration.

Additionally, the PSA results are used to demonstrate that a balanced design has been achieved for the UK ABWR, in which there are no single risks that make disproportionate contributions to the overall plant risk. This has included identification of any remaining vulnerabilities for consideration of further risk reduction measures.

The system modelling is based on the UK ABWR design drawings where these are available within the GDA timescales. Where detailed design information for the UK ABWR is not yet available then assumptions have been made based on general ABWR experience. All such assumptions have been fully documented.

In conclusion, the PSA for GDA demonstrates that the generic design of the UK ABWR reduces risks to levels that are ALARP, as far as is possible in the GDA stage. It is judged that the detailed site-specific design of the UK ABWR can be demonstrated to be ALARP by updated site-specific PSA modelling that is based on this GDA PSA. Development of the site-specific PSA will be the responsibility of any future licensee.

25.1 Introduction

Probabilistic Safety Assessment (PSA) is a key tool to: (i) assess the plant risks, (ii) identify potential plant vulnerabilities for consideration of further safety improvements and (iii) quantify the risks to the public. This document (PCSR Chapter 25) provides a summary of the final results for the UK ABWR GDA PSA, reflecting the GDA design, and the status of the quantification methodology. This document (PCSR Chapter 25) is based on the PSA Summary Report [Ref-25.1].

25.1.1 Background

Hitachi-GE has considerable experience in the development of PSAs for Boiling Water Reactors (BWRs) and Advanced Boiling Water Reactors (ABWRs) for applications during plant design, construction, and operation phases. Based on this experience, Hitachi-GE has developed a PSA for the Hitachi-GE standard ABWR design, referred to as the Hitachi-GE standard ABWR PSA [Ref-25.2].

The Hitachi-GE standard ABWR PSA was initially based on data and assumptions specific to existing operating plants. Although not fully applicable for meeting UK regulatory expectations of the Generic Design Assessment, the PSA has undergone a series of changes to meet those expectations. Initially, the UK ABWR PSA was adapted from the Hitachi-GE standard ABWR PSA, and the methodology for the UK ABWR PSA was defined in STEP 2 [Ref-25.3] [Ref-25.4] [Ref-25.5]. In order to provide early indication that the UK ABWR design would meet the risk targets defined in Hitachi-GE's Nuclear Safety and Environmental Design Principles (NSEDPs) [Ref-25.58] and ONR's Safety Assessment Principles (SAPs), a rebaselined risk profile was provided through sensitivity analysis based on the Hitachi-GE standard ABWR PSA.

In 2014, Hitachi-GE performed the initial quantification of internal events Level 1 PSA [Ref-25.6] and Level 2 PSA [Ref-25.129] for at power operations. During the GDA process, the modelling scope and level of detail have expanded progressively in order to meet UK regulatory expectations and in response to regulatory assessment comments. As examples of Internal Events at Power Level 1 and Level 2 PSAs, significant and notable submissions were made in December 2013 [Ref-25.2], December 2014 [Ref-25.6] [Ref-25.129], September 2015 [Ref-25.7], January 2016 [Ref-25.8] [Ref-25.74], June 2016 [Ref-25.9] [Ref-25.130] and April 2017 [Ref-25.10].

25.1.2 Purpose of this document

This document (PCSR Chapter 25) provides an overview of the role of PSA relevant to the safety case, and a summary of the results of the PSA for GDA. The technical details supporting this summary are provided by a number of Topic Reports. A tabulated summary of the Topic Reports pertinent to the sub-section headings in this document is provided in Appendix A .

25.1.3 Document structure

General descriptions of the overall approach and methodology commonly adopted for the Level 1 PSA, Level 2 PSA and Level 3 PSA for the different fault groups are provided in Sections 25.4, 25.5 and 25.6 respectively. For each fault group, a summary is provided on the following elements:

- results of the quantification,
- key assumptions made in the analysis,
- uncertainty/sensitivity analysis to provide meaningful interpretation of the results of quantification, and

- insights derived from assessment.

The entire consideration of these aspects within an integrated PSA framework is illustrated by the assessment for Internal Events at Power and this is summarised in Sections 25.4 to 25.7. In the assessment of other fault groups summarised in Sections 25.8 to 25.13, a similar reporting structure is adopted including a description of approach and methods that are specific for the analysis of individual fault groups. Section 25.15 provides a summary of the integrated results for all the fault groups and further insights derived from the integration.

The chapter is an integral part of the total assessment of faults and has links to a number of other PCSR chapters. The main links of this chapter with other GDA PCSR chapters are as follows.

- Chapter 2: Generic Site Envelope
  - The external hazard prioritisation for the PSA considers the characteristics of external hazards defined in Chapter 2.
- Chapter 5: General Design Aspects
  - The Plant Operational States (POSs) for the Shutdown PSA and SFP PSA refer to those defined in Chapter 5.
  - The safety functions relevant to the PSA are selected from the high level safety functions defined in Chapter 5.
- Chapter 6: External Hazards
  - The external hazard prioritisation for the PSA considers the external hazards identified and grouped in Chapter 6.
  - The external hazard prioritisation for the PSA considers the hazard definitions and general protection presented in Chapter 6.
- Chapter 7: Internal Hazards
  - The internal hazard prioritisation considers the internal hazards identified and grouped in Chapter 7.
  - The internal hazard PSAs consider the claims, arguments and deterministic safety evaluations in Chapter 7.
  - The initiating event frequency for the Turbine Missile PSA refers to the frequency of turbine disintegration hazard evaluated in Chapter 7.
- Chapter 9: General Description of the Unit (Facility)
  - The Global Plant Analysis Boundary (GPAB) for the Internal Fire PSA and Internal Flooding PSA refers to the generic site plan defined in Chapter 9.
- Chapter 10: Civil Works and Structures
  - The internal hazard prioritisation and external hazard prioritisation for the PSA consider the claims and design principles for civil structures presented in Chapter 10.
  - The internal hazard PSAs and external hazard PSAs consider the claims and design principles for civil structures presented in Chapter 10.
- Chapter 11: Reactor Core
  - The operation cycle length of 18 month for the PSA refers to Chapter 11.
- Chapter 12: Reactor Coolant Systems, Reactivity Control Systems and Associated Systems
  - The design of reactor coolant systems, reactivity control systems and associated systems modelled in the PSA refers to Chapter 12.

- Chapter 13: Engineered Safety Feature
  - The designs of emergency core cooling system and containment system modelled in the PSA refer to Chapter 13.
- Chapter 14: Reactor Control and Instrumentation
  - The design of Control and Instrumentation system modelled in the PSA refers to Chapter 14.
- Chapter 15: Electrical Power Supplies
  - The design of electrical power supply systems modelled in the PSA refers to Chapter 15.
- Chapter 16: Auxiliary Systems
  - The design of various support systems modelled in the PSA refers to Chapter 16.
  - The design of various severe accident mitigation systems modelled in the PSA refers to Chapter 16.
- Chapter 17: Steam and Power Conversion Systems
  - The design of steam and power conversion systems modelled in the PSA refers to Chapter 17.
- Chapter 19: Fuel Storage and Handling
  - The design of fuel handling machine, reactor building overhead crane and the information of lifting operations for the Fuel Route PSA, Shutdown PSA and SFP PSA refer to Chapter 19.
- Chapter 21: Human-Machine Interface
  - The design of Human-Machine Interface modelled for the PSA (including the Main Control Room analysis for Internal Fire PSA) refers to Chapter 21.
- Chapter 22: Emergency Preparedness
  - The procedures and guidelines discussed in Chapter 22 are considered for the success criteria determination and event sequence modelling in the PSA.
- Chapter 24: Design Basis Analysis
  - The identification of initiating events for the PSA (including non-reactor faults) refers to the Failure Modes and Effects Analysis (FMEA) in Chapter 24.
  - The computer codes used for the supporting thermal-hydraulic analyses are introduced in Chapter 24.
  - Some of the PSA success criteria are supported by the deterministic analyses in Chapter 24.
- Chapter 26: Beyond Design Basis and Severe Accident Analysis
  - PCV failure probability of the phenomenological event is evaluated in Chapter 26.
  - Severe accident analysis for faults at power, shutdown and SFP is performed in Chapter 26.
  - Source term analysis for faults at power, shutdown and SFP is performed in Chapter 26.
- Chapter 27: Human Factors
  - Human Reliability Analysis (HRA) gives the feasibility of a human action and the Human Error Probability (HEP).
- Chapter 28: ALARP Evaluation
  - PSA ALARP evaluation in Section 25.16 is integrated with other ALARP evaluations in Chapter 28.

25.2 Purpose and Scope of PSA

25.2.1 Purpose of PSA

PSA provides an integrated and structured analysis that combines engineering design and operational features in a consistent framework used to assess the plant risks, to identify potential plant vulnerabilities and to quantify the public risks. The PSA should be capable of providing the results necessary for comparison against relevant safety criteria. At a high level, quantitative results from PSA are often used to demonstrate compliance with safety goals or criteria, which are usually formulated in terms of quantitative estimates of core damage frequency, frequencies of radioactive releases of different types and societal risks. This may require the development of Level 1 PSA, Level 2 PSA or Level 3 PSA respectively. The objectives of the UK ABWR GDA PSA are;

- To quantify the overall risks represented by the facility to allow comparisons to be made against the UK Risk Targets (i.e. Targets 5, 6, 7, 8 and 9) as defined in the NSEDPs/SAPs, and other risk metrics used internationally, including Core Damage Frequency and Large Early Release Frequency,
- To assess and document the strengths and weaknesses of the design,
- To support the evaluations of potential modifications to the plant or improvements in operating conditions as part of ALARP demonstration, and
- To support other applications of safety decision making.

For the UK ABWR GDA, PSA is mainly used to demonstrate that the generic design meets the risk targets in NSEDPs [Ref-25.58] (equivalent to Targets 7, 8, and 9 of the SAPs) and to inform the design organizations of potential vulnerabilities to guide any future improvement as a part of the ALARP demonstration. To fulfil the above objectives, the scope is outlined in the next sub-section and the approach taken is outlined in Section 25.3.

25.2.2 Scope of PSA

PSA for the UK ABWR GDA should cover all significant sources of radioactivity and all relevant initiating faults identified at the facility. The scope is comprised of three elements: (i) sources of radioactivity at the facility, (ii) types of initiating events (internal events, internal hazards and external hazards) and (iii) operational modes. These elements and described as follows:

(1) All sources of radioactivity at the facility

Sources of radioactivity are identified as the fuel in the reactor, fuel in the Spent Fuel Pool (SFP), spent fuel in fuel route, spent fuel in the Interim Storage Facility, sources in the Main Turbine and radioactive waste system, and irradiated structures and materials.

(2) All types of initiating faults

Internal events are identified in a systematic manner, including logic tree analysis and review of existing analysis for each source of radioactivity. Design Basis (DB) initiating faults [Chapter 24 of PCSR Rev. C], which affect the safety of the facility, are identified and treated through the process of, internal events, Internal Hazard and External Hazard analysis. In the PSA, each of these initiating faults is reviewed as the starting point, and hazards whose risk should be quantified by PSA are identified through the hazard identification process.

(3) All operational modes

In the analysis for fuel in the reactor, the at power operation mode and all shutdown modes are taken into account. The low power mode is also addressed.

The full scope PSA for design assessment is substantiated through the process of identification, grouping and prioritisation of all sources of radioactivity, types of initiating faults and operational modes. A systematic process of initiating event identification, internal hazard identification/prioritisation and external hazard identification/prioritisation is important to achieve a full scope PSA for this design assessment.

After the initiating events are identified, the analysis is performed using a progressive prioritization approach. This approach ensures that the depth of detail in the PSA is appropriate for the assessment of risk at each stage in the licensing process.

The basic principles for consideration in addressing the issue of depth are that:

(1) The PSA cannot have greater depth of details than the design of the UK ABWR SSCs during GDA, which is at the level of specification of requirements prior to detailed design in the site specific phase of the project;

(2) The PSA reflects the Design Reference, and

(3) All internal and external hazards are considered in the PSA, but the PSA focuses on those events which are risk significant.

When the level of plant design detail is insufficient for the PSA model, appropriate assumptions are developed to allow completion of the PSA. The list of PSA assumptions relevant to all PSA aspects was developed. This list of assumptions has been reviewed and updated throughout GDA as new information became available and changes occurred. Additionally, the PSA and risk insights are used to support the ALARP assessment and shared with the appropriate design teams in accordance with Hitachi-GE's design process.

For links to the Generic Environmental Permit (GEP) and the Conceptual Security Arrangements (CSA) documentation, please refer to PCSR Chapter 1: Introduction. For the GEP, where specific references are required, for example in Radioactive Waste Management, Radiation Protection, and Decommissioning, they are included in specific sections within the PCSR.

25.3 PSA Study Approach for UK ABWR PSA

25.3.1 Approach

This document demonstrates how the PSA model provides an adequate representation of the UK ABWR. Up-to-date design information and operational features for the UK ABWR GDA are applied to the PSA model.

For each element of PSA, tasks are defined to support development of the PSA. The following types of documents are developed for each task:

- Methodology report,
- Task procedure,
- Task analysis files,
- Documentation data base, and
- Summary report

The methodology report describes the basic approach for developing the PSA model for each PSA task. The task procedure describes rules for developing the PSA model, which is mainly for internal use. Supporting documents summarise the results of individual tasks, e.g. initiating event analysis, system analysis. Task analysis files store calculation history, and contains the information needed to develop the model and to perform calculations. The document data base is a set of documents on references and assumptions used for developing the PSA model. The Summary report provides a summary of both the methodology and results for each task.

Design input which is necessary for PSA modelling was defined or assumed through the activities in the UK ABWR PSA. System modelling is based on the UK ABWR design drawings and/or assumptions (if detailed design information for the UK ABWR is not yet available). Before PSA quantification, system modelling details and assumptions were reviewed by the relevant engineering sections.

The PSA model for each task is quantified using CAFTA software.

25.3.2 Requirements and Methodology

PSA methodologies were prepared for Level 1 PSA [Ref-25.3], Level 2 PSA [Ref-25.3] and Level 3 PSA [Ref-25.5]. Methodologies for other scope elements are described for each analysis. Methodology reports for the internal events shutdown PSA, SFP PSA, internal fire PSA, internal flooding PSA and seismic PSA were prepared during Step 3 and Step 4.

To perform the GDA PSA, the appropriate models and data for the generic design phase was used for the UK ABWR PSA. When the level of plant design detail is insufficient for the PSA, appropriate assumptions were developed to allow completion of the PSA.

25.3.3 Application of PSA

Hitachi-GE envisioned the use of PSA in a number of GDA activities. This section describes Hitachi-GE’s approach on the application of PSA in GDA.

In general, the PSA provides an input to support the following activities:

(1) Designing the facilities,

(2) Supporting modifications to the design and operation,

(3) Supporting the demonstration that offsite risks are tolerable and ALARP,

(4) Informing the selection of safety function categories and the safety classifications of structures, systems and components,

(5) Setting operating rules,

(6) Informing arrangements for examination, maintenance inspection and testing (e.g. the maintenance frequencies for these activities),

(7) Plant configuration control (including maintenance planning), which for power reactors is normally through the use of risk monitors,

(8) Event analysis and investigating significant incidents and events,

(9) Developing and changing operating procedures and associated training programmes for managing faults and accidents (including severe accidents),

(10) Helping to determine initiating fault frequencies for Design Basis Accidents, and

(11) Providing an input to Severe Accident Analyses and to analyses supporting the Radiation Emergency Preparedness and Public Information Regulations 2001, (REPPIR).

It should be recognised that the emphasis of the application will change from GDA to other phases in the life cycle of a UK ABWR plant. Among the listed applications, items a) to c) were more relevant in GDA. Use of PSA in ALARP assessment is presented in Section 25.16. In addition to that, PSA system analysis results were utilised for the comparison with the Safety Case Development Manual (SCDM) reliability target for a part of systems, which is determined by safety class of the system. Furthermore, PSA identifies the Human Failure Events (HFEs), which are used as one of the inputs for Human-Based Safety Claim [Ref-25.104]. The PSA applications will be expanded at the post GDA phase.

25.3.4 Use of PSA to support Potential Design Improvements

25.3.4.1 Objective of this process

The PSA can provide insights from both the base results and the associated sensitivity and uncertainty studies, which supports the ALARP assessment in the UK ABWR Safety Case. 'Insights' include identifying the plant failures involved in the largest or highest frequency releases. These could be initiating events or safety systems. The following are key aspects regarding the use of PSA to inform the design process.

25.3.4.2 Overall process and related procedure

The PSA is used as a tool to support the design development. In this process, the PSA engineers develop the PSA insights and recommend design enhancements based on these insights. To provide the inputs to the design and/or development, PSA engineers have performed the following steps:

- Identification of insights,
- Communication of insights and disposition of candidate design enhancements [Ref-25.11], and
- Preparation of report to support the ALARP claim

25.3.5 PSA Development

The PSA for the UK ABWR is developed through the Hitachi-GE design process. The design process includes the use of the Hitachi-GE quality assessment process, which is applied to the PSA.

System modelling is based on the design drawings. Assumptions are made as necessary for PSA development. Before PSA quantification, system modelling details and assumptions are reviewed by relevant engineering sections (including the human factor team). The design information of UK ABWR available in GDA are:

- P&ID (Piping &Instrumentation Diagram),
- Single Line Diagram (for M/C and P/C),
- SDD (System Design Description), and
- Basis of Safety Cases for some systems.

Specific references of design documents for each system analysis are described in the relevant PSA documents.

25.3.6 Design Reference for PSA

The PSA was developed to support the GDA process. The design information was gathered during the development of the PSA models. If design information was not available at that stage of the design process, design assumptions were developed. To capture the assumptions and to confirm the fidelity of GDA project, a base assumption review and updates of the Task Analysis Files were performed late in GDA Step 3.

The PSA support team (Engineering section members) reviewed the PSA documents to reflect the UK ABWR design at DRP (Design Reference Point) in advance. The review results were reflected in the revised PSA model. If the design was not changed during the review period, the information on this is recorded in the relevant document and Hitachi-GE's IT system.

As a result of system analyses, the design changes in Batches 1 to 3 were incorporated in the UK ABWR PSA. The design changes after DRP including Batches 4 and 5 were reviewed after the development of the UK ABWR PSA as described in Section 25.3.8.3.

25.3.7 PSA Assumptions

Assumptions for the UK ABWR PSA were made and documented in order to capture all the PSA assumptions related to the design, procedures, acceptable limits, success criteria, accident sequence, and operator actions. The list of assumptions was developed and periodically updated such that:

- All assumptions made throughout the study are clearly identified, described and properly justified.
- The specific aspects of the PSA models or data related to these assumptions are clearly identified.
- All assumptions are captured from the appropriate documents systematically.

The review of PSA assumptions will be needed if the PSA is updated, which is managed by the commitment log for PSA at the post-GDA phase.

25.3.8 Quality Assurance and PSA updating Process

25.3.8.1 PSA Quality Assurance Plan

The Quality Management Plan (QMP) has been established to describe the process for quality assurance, environmental and safety management activities during the GDA of the UK ABWR which are coordinated and delivered by the Requesting Party; Hitachi-GE Nuclear Energy, Ltd. in accordance with "ISO 90001-2008", "ISO 14001-2004" and IAEA Safety Requirement No. GS-R-3.

The purpose of the quality assurance process for GDA is to ensure nuclear safety as a first priority, and to ensure that there is conformity to all applicable laws, codes, standards, regulations and UK Licensing and Permitting Conditions, to fulfil regulatory expectations.

The Quality Assurance and management system arrangements for the GDA [Chapter 4: Safety Management throughout Plant Lifecycle] were established at the end of Step 1. The QMP includes the arrangements for addressing;

- Safety culture,
- Graded approach,
- Management responsibility,
- Resource management,
- Process implementation including design and document control, and
- Measurement, assessment and improvement.

A number of supporting procedures to the QMP have been established. QMP describes how the production of important safety case documents like the PCSR are controlled, reviewed, independently verified and approved. "SQEP Requirements for Hitachi-GE and Supplier Personnel" specifies the development of role profiles and competency evaluations to ensure that Suitably Qualified and Experienced Persons are responsible for the delivery of the GDA project. The QA arrangements for the GDA and the production of the PCSR will ensure that the UK ABWR will be built as designed. These documents and records for the GDA will be controlled in accordance with control of general documents and records.

25.3.8.2 External peer review process

Apart from the Hitachi-GE internal review process, a specific feature in the development of UK ABWR PSA is the implementation of external peer reviews by independent international PSA experts. This is to ensure the PSA is of sufficient technical adequacy to support the GDA and is consistent with the demands of modern standard and good practice for PSA development and performance. Since August 2015, the following peer reviews have been performed:

- Level 1 PSA,
- Low Power and Shutdown PSA,
- Spent Fuel Pool/Heavy Loads PSA,
- Seismic PSA,
- PSA Hazards Prioritisation,
- Level 2 PSA,
- PSA Sensitivity and Uncertainty Analysis,
- Fuel Route PSA,
- Other Hazards PSA,
- Level 3 PSA,

- Fire PSA, and
- Internal Flood PSA.

The peer reviews listed above were performed using the NEI 05-04 [Ref-25.12] process and the ASME/ANS PRA Standards [Ref-25.13]. The peer reviews also evaluated against the requirements/expectations of the ONR assessment guidance - Nuclear Safety Technical Assessment Guide (TAG) NS-TAST-GD-030, Revision 5.

Since the performance of the specific peer review in GDA:

- PSA model and documentation enhancements have been implemented based on the Facts and Observations (F&Os) generated during the specific peer reviews.
- Over 70 percent of the F&Os have been closed and the outstanding items are recorded with the resolution to be performed in the future.

25.3.8.3 PSA updating Process

For the update of the UK ABWR PSA, Hitachi-GE prepared an internal procedure to assess the change of models in GDA. In this procedure, impact of changes and the necessity of changes are evaluated and recorded when PSA changes are identified. The design changes after DRP including Batches 4 and 5 are reviewed after the development of the UK ABWR PSA. The impact of changes and the necessity of changes are summarised in “Model Change Tracking/ Risk Impact Evaluation (MCT/RIE) Database”. It is confirmed that the design changes after DRP do not impact significantly on PSA results and PSA insights.

25.4 Level 1 PSA for Internal Events at Power

This section covers the scope of Internal Events at Power (IEAP) considered in the Level 1 PSA, and presents the methodology and results. Results are presented in terms of Core Damage Frequency (CDF) per year. The same methodology was applied for other fault groups unless otherwise stated in the section below.

The probabilistic safety objectives adopted for the UK ABWR are stated in Section 25.15.

In the PSA, event trees are used for estimating the CDF due to each Initiating Event (IE). Fault trees are used to estimate the failure probability of the system missions following each of the IEs.

IEs are identified and bounded (see Section 25.4.1). Event trees are developed based on the associated success criteria for each IE and corresponding event sequence analysis (see Sections 25.4.2 and 25.4.3). Note that this section is limited to internal events only. The contributions from internal and external hazards are addressed in Sections 25.10 and 25.11.

Fault trees are developed for all systems required in the mitigation of the IE (see Section 25.4.4).

The event tree and fault tree analyses include human reliability analysis (see Section 25.4.5), data analysis (see section 25.4.6) and common cause failure analysis (see Section 25.4.7).

The risk quantification is carried out using CAFTA [Ref-25.32]. This software enables integrated event tree and fault tree modelling. The quantification methodology is presented in Section 25.4.8 and results are presented in Section 25.7.

25.4.1 Identification and Grouping of Initiating Event

25.4.1.1 Scope

IAEA-TECDOC-1511 [Ref-25.14] provides the definition of IE: an event which could directly lead to core damage or challenges normal plant operation and requires successful mitigation to prevent core damage. IEs identified typically include transients of various types, Loss of Coolant Accidents (LOCAs) and Support System Initiating Events (SSIEs). IEs under low power conditions are also identified based on Failure Modes and Effects Analysis (FMEA) exercise, and the identified IEs are included in IEs at Power.

In addition, planned and unplanned manual shutdowns which seldom place demands on any standby safety equipment are treated as IEs because of their high frequency and because they represent changes in operating states which result in the demand on available equipment to reach a safe shutdown condition.

This section summarises the IE analysis performed for the Internal Events At Power (IEAP) Level 1 PSA in the UK ABWR. The selection of initiating events to be modelled in a PSA is the first essential step in the development of a plant PSA model. Before the event sequences can be defined and analysed, the identified initiating events must be divided into appropriate groups with similar characteristics in order to efficiently model the potential failure scenarios.

The main objectives of IE analysis for IEAP Level 1 PSA are:

- To identify a reasonably complete set of the events that interrupt normal plant operation and that require successful mitigation to prevent core damage, so that no significant contributor to core damage is omitted (see Section 25.4.1.2),

- To group initiating events to facilitate the efficient modelling of the associated plant response in order to effectively model the propagation of the resulting accident sequences (events included in the same group have similar mitigation requirements, or are bounded by the limiting mitigation requirements for the 'representative initiating event' for the group) (see Section 25.4.1.3), and

- To provide estimates for the frequencies of the initiating event groups using information available and associated estimation techniques (see Section 25.4.1.4).

The scope of this section is limited to internal events occurring whilst the reactor at Power.

The list of initiating events will be updated during the post GDA phase of the UK ABWR as the number of events that need to be modelled in the PSA will be increased.

25.4.1.2 Identification of Initiating Events during Full and Low Power Operation

SSIEs are defined as unplanned normal shutdown resulting from loss of a particular support system (with dependent failures). The following steps were taken to identify all possible IEs:

Step 1: Review of lists of IEs from PSAs for similar units

Lists of IEs developed in PSAs for similar units (Hitachi-GE Standard ABWR [Ref-25.2] and Lungmen ABWR [Ref-25.39] were reviewed in order to identify potential IEs applicable to the investigated unit.

Step 2: Review of generic lists of IEs

EPRI/NP-2230 [Ref-25.15] was selected as the generic list of IEs for review because it has been widely referenced by industry PSAs in US and Japan as the generic source. NUREG/CR-5750 [Ref-25.127] was also reviewed because it provides the categorisation of IEs for the operational experience in the US.

Step 3: Review of operational experience

Operational experience of UK ABWR is not yet available. Those of similar units, BWR/ABWR plants in Japan and BWR plants in the US, were reviewed in order to identify events that happened in the past.

Once the UK ABWR is in operation, the operational experience will be factored into the IE identification.

Step 4: Review of IE lists for Fault Assessment

The latest available IE list in the DBA Fault Assessment [Chapter 24 of PCSR Rev. C] was reviewed. The IE list was updated based on the FMEA exercises (see Step 5).

Step 5: Inductive analysis

It is recognised that since the preceding steps for IE identification focus on external sources and domestic experience, there remains the risk of missing IE candidates unique to UK ABWR. Therefore, a systematic evaluation of each system of the UK ABWR has been performed to assess the possibility of an IE occurring due to a failure of the system. All systems whose failure would result in a disturbance in the normal plant operation have been reviewed using the FMEA approach. The systems considered include front-line and support systems. In the FMEA, individual component failures are considered as well as Common Cause Failure (CCF) and human error events.

Loss of Coolant Accident (LOCA) was also considered in each system, as highlighted in the following paragraphs.

LOCA Inside Containment IEs:

The reactor vessel drawings were reviewed to identify all penetrations and RPV vessel locations that could be susceptible to a LOCA to ensure that the identified LOCA initiating events are complete.

For the LOCA initiators, IE candidates were identified by location, and the PSA includes location specific LOCA initiators considering the following items:

- The loss of injection path from the affected loop of Emergency Core Cooling System (ECCS),
- The difference in success criteria for long term core cooling,
- The effect of the LOCA on causing release of debris into the suppression pool,
- The adverse effects on mitigation capability due to a LOCA below Top of Active Fuel, and
- All penetrations and RPV vessel locations that could be susceptible to a LOCA.

Interfacing System Loss of Coolant Accident (ISLOCA) IEs:

For the ISLOCA initiators, IE candidates were identified by reviewing the steam and liquid lines (normally isolated) connected to the RPV based on NSAC 154 [Ref-25.16].

Break Outside Containment (BOC) IEs:

For the BOC initiators, IE candidates were identified by reviewing the steam and liquid lines (normally not isolated) connected to the RPV. Lines that are isolated at the containment boundary by a normally closed Motor Operated Containment Isolation Valve (CIV) are not included in the BOC analysis.

Low Power IEs:

An analysis was performed to identify initiators unique to low power operations and initiators applicable to full power operations that might be impacted by low power conditions. Initiators unique to low power operations were reviewed via discussions with personnel experienced in ABWR and BWR operations. As a result, low power conditions are found to introduce no significant initiators unique to low power and are found not to have a significant impact on the results of the UK ABWR IEAP PSA.

Step 6: Listing of IE candidates

The list of potential initiating events, presented within broad categories (e.g. anticipated transients, LOCAs), was developed based on the results of the analyses performed within Step 1 to Step 5 described above. Those IE candidates that were identified in multiple steps were combined together.

Step 7: Dependency identification

Complete and partial dependencies between the identified potential IEs and the post initiator functions (in both front-line and support systems) were systematically identified by reviewing the definitions of IEs, essential support systems of credit functions, RPV injection lines and so on. The objective of the dependency identification in this step was to support IE screening and IE grouping in the following subtasks.

25.4.1.3 Grouping of Initiating Events

Step 1: Grouping of IEs

Multiple IEs are grouped in a single group only when the following can be assured:

- Events have the same safe and unsafe end states and lead to similar accident progression in terms of plant response, success criteria, timing, and the effect on the operability of relevant mitigating systems and operators performance, or

- Events can be subsumed into a group and bounded by the worst case impacts within the 'new' group.

Each IE group is clearly defined and characterised. The information provided is sufficient for the quantification of IE frequencies (i.e. its causes are identified) and for the development of accident sequence models (i.e. its impact on plant is stated).

Step 2: Listing of IE groups

A list of IE groups has been compiled as shown in Table 25.5.1-1. A representative or hypothetical (bounding) IE for further modelling of each IE group has been selected. The most conservative features in terms of accident progression and success criteria of an IE included in the group have been assigned for the representative or hypothetical (bounding) IE.

Step 3: Impact of IE groups on systems

Dependencies between the grouped initiators and the post initiator functions (front-line and support systems) have been systematically identified, and a dependency matrix has been developed. System fault trees and functional fault trees have been used as part of this review where necessary (especially for SSIEs).

The impact of each IE group on the frontline systems and support systems have been summarised based on the outcome of the Dependency Identification (Step 7 of the previous subsection). The objective of this step was to support the success criteria determination and accident sequence analysis (event tree development).

25.4.1.4 Quantification of Initiating Events Frequency

In this subtask, mean frequencies (per calendar year) of the IE groups are addressed. Note that many frequencies are presented as failures per reactor year. In order to convert from reactor year to calendar year, those frequencies are multiplied by the fraction of year that, on average, the plant is at Power which is taken to be 0.9. This average is assumed based on the experience of the US industry-average presented in NUREG/CR-6928 [Ref-25.17]. Note that the fault group frequency is the sum of the individual faults grouped, and the group is represented by the most onerous fault in the group.

Step 1: Manual shutdown group

In this step, frequencies of the manual shutdown group (including the Technical Specifications (TS) manual shutdown group) are addressed.

Manual shutdown group except TS manual shutdown group

Manual shutdown group (w/o manual scram) includes planned and unplanned manual shutdowns not impacting any credited mitigation function. This group do not fit the IE definition in the US generic data and hence the frequency is not available there. Therefore, the frequency of operating ABWR plants in Japan, i.e. 1.0 /ry, is conservatively applied for UK ABWR PSA.

Technical Specification initiator group

The development of SSIEs has been integrated into the same set of initiating event fault trees. The use of SSIE fault trees allows the integration of TS initiating events; however the treatment of the failure of standby equipment that results in an unplanned administrative shutdown has been developed differently than for normally running components. The standby equipment unavailability is detected during periodic testing of the equipment and the number of tests per year has been considered in the calculation of the initiating event frequency of the respective equipment.

Step 2: Anticipated transient groups except loss of offsite power

General transient group

The mean frequency of the “General Transient (BWR)” category in the latest available data (2010 Parameter Estimation Update) of NUREG/CR-6928 is 7.62E-1 (/ry) (in 1997-2010 as the optimised baseline period). The mean frequency of “Very small LOCA/leak (BWR)” is 4.36E-3 (/ry) (in 1992-2010 as the optimised baseline period) [Ref-25.17].

The sum of those frequencies is 7.66E-1 (/ry).

Loss of condenser heat sink group

The mean frequency of “Loss of Condenser Heat Sink (BWR)” category in the latest available data (2010 Parameter Estimation Update) of NUREG/CR-6928 is 1.39E-1 (/ry) (in 1996-2010 as the optimised baseline period).

Loss of feedwater group

The mean frequency of “Loss of Main Feedwater” category in the latest available data (2010 Parameter Estimation Update) of NUREG/CR-6928 is 6.89E-2 (/ry) (in 1993-2010 as the optimised baseline period).

Inadvertent open relief valve group

The mean frequency of “Stuck Open Safety/Relief Valve-1 (BWR)” category in the latest available data (2010 Parameter Estimation Update) of NUREG/CR-6928 is 1.63E-2 (/ry) (in 1993-2010 as the optimised baseline period).

Inadvertent ADS actuation

A fault tree is developed to quantify the frequency of this IE. Spurious signal of low reactor water level (L1) is considered for Transient ADS actuation. All the relevant transmitters are considered.

PCV pressurisation by leakage of nitrogen gas

A fault tree is developed to quantify the frequency of this IE. External leak (large) of relevant components are considered.

Step 3: Loss of offsite power

Frequencies of LOOP IEs are divided in the following 16 categories.

- Grid Related Loss of offsite power (< 0.5 h),
- Grid Related Loss of offsite power (0.5 to 8 h),
- Grid Related Loss of offsite power (8 to 14 h),
- Grid Related Loss of offsite power (> 14 h),
- Plant Centred Loss of offsite power (< 0.5 h),
- Plant Centred Loss of offsite power (0.5 to 8 h),
- Plant Centred Loss of offsite power (8 to 14 h),
- Plant Centred Loss of offsite power (> 14 h),
- Switchyard Centred Loss of offsite power (< 0.5 h),
- Switchyard Centred Loss of offsite power (0.5 to 8 h),
- Switchyard Centred Loss of offsite power (8 to 14 h),
- Switchyard Centred Loss of offsite power (> 14 h),
- Weather Related Loss of offsite power (< 0.5 h),
- Weather Related Loss of offsite power (0.5 to 8 h),
- Weather Related Loss of offsite power (8 to 14 h), and
- Weather Related Loss of offsite power (> 14 h).

The Plant Centred, Switchyard Centred and Weather Related LOOP categories were calculated based on the occurrence frequencies and duration curve fits in the US [Ref-25.43] (NRC website, “Analysis of Loss-of-Offsite-Power Events 1998 to 2013”). The Grid Related LOOP categories were calculated based on an existing study in the UK (for occurrence frequency) and the US LOOP data (for LOOP duration curve fit).

Consequential LOOP is treated not as an initiating event but as a conditional probability in the branch of each LOCA and Transient based (non-LOCA) event tree. The conditional LOOP probabilities are distinguished between non-LOCA conditions and LOCA conditions.

The probability of a conditional LOOP occurring immediately following a transient and a LOCA is estimated based on NUREG/CR-6890 [Ref-25.41].

Step 4: Support system initiators

The configuration of the SSIE is based on the multiplier method described in the Electric Power Research Institute (EPRI) document [Ref-25.19]. The multiplier method is based on using the existing (mitigating) system fault trees as the basis for the initiating event fault tree. The SSIE fault trees have been developed such that maximum alignment has been retained between the initiating event fault trees and the mitigating system fault trees. The SSIE fault trees contain a similar structure to the mitigating system "fails to run" fault trees (including auto-backup of standby trains if appropriate). The conversion from the probability of failure over a 24 hours mission period within the (post initiating event) system fault tree to an annual frequency is accomplished using a 365 days "multiplier" basic event (IE-XXXX) and plant unavailability "0.9 multiplier" basic event towards the top of the fault tree logic. The advantage of this method is that it allows modelling of the multiple trains or divisions in the initiating event fault trees and can also include common cause failures. Furthermore, as CAFTA incorporates all sequences into a master fault tree model as part of the quantification process, the initiator to mitigation system dependencies are automatically captured: The failure events which appears in the SSIE fault trees (including potential CCF events) impacts the mitigated systems if they also appear in the mitigating system fault trees.

Step 5: LOCAs within containment

Frequencies of LOCA within containment are obtained, considering three LOCA categories; Large LOCA, Medium LOCA and Small LOCA. To determine the frequency associated with each ABWR Reactor Coolant Pressure Boundary (RCPB) line, each NUREG/CR-6928 [Ref-25.17] LOCA frequency is apportioned to the associated LOCA Lines based proportionally on their length.

Step 6: Excessive LOCA

The estimated frequency of 1E-8 is taken from the Category 6 (End-of-Plant-License Estimate for BWR) of NUREG-1829 [Ref-25.20] and corresponds to "a catastrophic rupture of the reactor pressure vessel" for BWR. It envelopes the frequency after 60 years of operation according to Table 4.4 of NUREG-1829. Due to such a small number and associated uncertainty this number is used as the frequency per critical year. This number is then multiplied by the fraction of year that, on average, the plant is at Power and the mean IE frequency of 9.0E-9 (/y (calendar year)) is applied.

Step 7: ISLOCAs

The calculated frequencies for interfacing-system LOCAs are calculated based on the methodology in NSAC-154 [Ref-25.16], which uses elements of NUREG/ CR-5124 [Ref-25.40].

Step 8: BOCs

The initiating event frequency of each of the identified Breaks Outside Containment (BOC) IEs are derived from the pipe break frequency per unit length contained in EPRI 300200079 [Ref-25.42].

Summary of IE frequencies

The IE frequencies based on the generic data (i.e., NUREG/CR-6928), expert elicitation process (i.e., NUREG-1829) and LOOP IEs are summarised in Table 25.4.1-2. The IE frequencies calculated by delicate fault trees (i.e., SSIEs, BOCs, ISLOCAs) are summarised in the PSA Summary Report [Ref-25.1].

Table 25.4.1-1 List of Initiating Events Included in the PSA (1/2)

Broad category	IE group	Event Tree
Anticipated Transients	General transient	TG
	Loss of condenser heat sink	TM
	Loss of feedwater	TF
	Inadvertent open relief valve	TI
	Inadvertent ADS actuation	A-ADS
	PCV pressurisation by leakage of nitrogen gas	TG*
	Grid Related Loss of offsite power (< 0.5 h)	TE1
	Plant Centred Loss of offsite power (< 0.5 h)	TE1
	Switchyard Centred Loss of offsite power (< 0.5 h)	TE1
	Weather Related Loss of offsite power (< 0.5 h)	TE1
	Grid Related Loss of offsite power (0.5 to 8 h)	TE2
	Plant Centred Loss of offsite power (0.5 to 8 h)	TE2
	Switchyard Centred Loss of offsite power (0.5 to 8 h)	TE2
	Weather Related Loss of offsite power (0.5 to 8 h)	TE2
	Grid Related Loss of offsite power (8 to 14 h)	TE3
	Plant Centred Loss of offsite power (8 to 14 h)	TE3
	Switchyard Centred Loss of offsite power (8 to 14 h)	TE3
	Weather Related Loss of offsite power (8 to 14 h)	TE3
	Grid Related Loss of offsite power (> 14 h)	TE4
	Plant Centred Loss of offsite power (> 14 h)	TE4
	Switchyard Centred Loss of offsite power (> 14 h)	TE4
	Weather Related Loss of offsite power (> 14 h)	TE4
LOCAs	Small LOCA (RHR-B and RPV Bottom Head Sample Line) within containment	S2-RHRB
	Small LOCA (RHR-C) within containment	S-2RHRC
	Small LOCA (Reactor Water Sampling Line A) within containment	S2-RWLA
	Small LOCA (Reactor Water Sampling Line B) within containment	S2-RWLB
	Small LOCA (Reactor Water Sampling Line C) within containment	S2-RWLC
	Small LOCA (Reactor Water Sampling Line D) within containment	S2-RWLD
	Small LOCA (SLC) within containment	S2-SLC
	Small LOCA (CUW Bottom Head Drain and Instrument Taps) within containment	S2
	Medium LOCA (HPCF-B) within containment	S1-HPCFB
	Medium LOCA (HPCF-C) within containment	S1-HPCFC
	Medium LOCA (Bottom Drain Line, RCIC and RPV Head Spray and Instrument Taps) within containment	S1-BDL
	Medium LOCA (RHR-B) within containment	S1-RHRB
	Medium LOCA (RHR-C) within containment	S1-RHRC
	Large LOCA (MSL, SRV Inlets, FW-B, RHR A, B, C Suction, CUW mid-vessel suction) within containment	A
	Large LOCA (FW-A) within containment	A-FWA
	Excessive LOCA	S4
	BOC – Main steam line break –	S3-BMSFB*

Table 25.4.1-1 List of Initiating Events Included in the PSA (2/2)

Broad category	IE group	Event Tree
LOCAs	BOC – FW-A line break –	S3-BFWA
	BOC – FW-B line break –	S3-BMSFB*
	BOC – RCIC line break –	S3-BRCIC
	BOC – Reactor Water Clean-up line break –	S3-BCUW
	BOC – SAM line break –	S3-BSAM
	ISLOCA –RHR-A suction line–	S3-IRA
	ISLOCA –RHR-B suction line–	S3-IRB
	ISLOCA –RHR-C suction line–	S3-IRC
	ISLOCA –HPCF-B injection line–	S3-IHLB*
	ISLOCA –LPFL-B injection line–	
	ISLOCA –HPCF-C injection line–	S3-IHLC*
	ISLOCA –LPFL-C injection line–	
Manual Shutdown	Manual shutdown	MS**
Support system initiators (SSIEs)	Manual shutdown (Tech Spec Initiator)	
	Loss of Class 1 AC	
	Loss of Class 3 AC	
	Loss of Class 1 DC	
	Loss of Class 2 DC	
	Loss of RCW/RSW	
	Loss of R/BEEE/Z HVAC	
	Loss of TBNEEE HVAC	
	Loss of T/B HVAC	
	Loss of Hx/B-N HVAC	
	Loss of CBEEE/Z (A) HVAC	
	Loss of CBEEE/Z (B) HVAC	
	Loss of CBEEE/Z (C) HVAC	
	Loss of MCR HVAC	
	Loss of HNCW	
	CCF of Class 1 controller	
	CCF of Class 3 controller	
	Loss of Instrument or Control Air System	

*Two IEs are analysed by the same event tree due to identical dependencies.

**All the manual shutdown IEs and SSIEs are analysed by the same event tree. Dependent failures are automatically captured in the one-top master fault tree.

Table 25.4.1-2 IE Frequency Summary (1/2)

Broad category	IE group	Mean frequency per calendar year	Distribution	Reference
Anticipated Transients	General transient	6.90E-1	21.0*	NUREG/CR-6928 (2010 update)
	Loss of condenser heat sink	1.26E-1	2.9*	
	Loss of feedwater	6.21E-2	2.2*	
	Inadvertent open relief valve	1.47E-2	2.5*	
	Inadvertent ADS actuation	7.57E-04	19.0**	Quantified by fault trees
	PCV pressurization by leakage of nitrogen gas	1.32E-02	9.8**	
	Grid Related Loss of offsite power (< 0.5 h)	1.93E-3	0.3*	NUREG/CR-6928 (2010 Update)
	Grid Related Loss of offsite power (0.5 to 8 h)	1.12E-2	0.3*	
	Grid Related Loss of offsite power (8 to 14 h)	1.07E-3	0.3*	
	Grid Related Loss of offsite power (> 14 h)	7.92E-4	0.3*	
	Plant Centred Loss of offsite power (< 0.5 h)	8.29E-4	1.0*	
	Plant Centred Loss of offsite power (0.5 to 8 h)	9.10E-4	1.0*	
	Plant Centred Loss of offsite power (8 to 14 h)	3.69E-5	1.0*	
	Plant Centred Loss of offsite power (> 14 h)	2.41E-5	1.0*	
	Switchyard Centred Loss of offsite power (< 0.5 h)	4.15E-3	0.5*	
	Switchyard Centred Loss of offsite power (0.5 to 8 h)	7.37E-3	0.5*	
	Switchyard Centred Loss of offsite power (8 to 14 h)	5.54E-4	0.5*	
	Switchyard Centred Loss of offsite power (> 14 h)	5.19E-4	0.5*	
	Weather Related Loss of offsite power (< 0.5 h)	7.77E-4	0.3*	
	Weather Related Loss of offsite power (0.5 to 8 h)	2.14E-3	0.3*	
	Weather Related Loss of offsite power (8 to 14 h)	4.19E-4	0.3*	
	Weather Related Loss of offsite power (> 14 h)	1.16E-3	0.3*	

* The EF in this case is the α factor as the distribution is the gamma type

** EF = SQRT (95 percentile / 5 percentile)

Table 25.4.1-2 IE Frequency Summary (2/2)

Broad category	IE group	Mean frequency per calendar year	Distribution	Reference
LOCAs	Small LOCA (RHR-B and RPV Bottom Head Sample Line) within containment	1.35E-5	8.4*	NUREG-1829
	Small LOCA (RHR-C) within containment	7.36E-7	6.7*	
	Small LOCA (Reactor Water Sampling Line A) within containment	3.61E-5	8.4*	
	Small LOCA (Reactor Water Sampling Line B) within containment	3.03E-5	8.4*	
	Small LOCA (Reactor Water Sampling Line C) within containment	4.04E-5	8.4*	
	Small LOCA (Reactor Water Sampling Line D) within containment	4.54E-5	8.4*	
	Small LOCA (SLC) within containment	8.58E-7	6.7*	
	Small LOCA (CUW Bottom Head Drain and Instrument Taps) within containment	8.00E-5	6.7*	
	Medium LOCA (HPCF-B) within containment	1.81E-6	6.7*	
	Medium LOCA (HPCF-C) within containment	1.81E-6	6.7*	
	Medium LOCA (RPV bottom drain line, RCIC and RPV Head Spray and Instrument Taps) within containment	2.87E-4	8.4*	
	Medium LOCA (RHR-B) within containment	1.61E-6	6.7*	
	Medium LOCA (RHR-C) within containment	1.61E-6	6.7*	
	Large LOCA (MSL, SRV Inlets, FW-B, RHR A, B, C Suction, CUW mid-vessel suction) within containment	7.35E-6	9.1*	
	Large LOCA (FW-A) within containment	1.38E-6	9.1*	
	Excessive LOCA (RPV rupture)	9.00E-9	62**	

* The EF in this case is the α factor as the distribution is the gamma type

** EF = SQRT (95 percentile / 5 percentile)

25.4.2 Determination of Success Criteria

Determination of success criteria is part of the event sequence analysis task. This section summarises the success criteria determination mainly based on IAEA-TECDOC-1511 [Ref-25.14] and AESJ standard [Ref-25.21].

The main objective in the success criteria determination for the IEAP Level 1 PSA is to determine for given initiating events what represents a successful or unsuccessful plant response and to translate this information into detailed plant system and operator action success criteria [Ref-25.14];

The subtasks in the success criteria determination are [Ref-25.14]:

- Subtask 1: Definition of Overall and Detailed Success Criteria, and
- Subtask 2: Thermal Hydraulic Analyses and other Assessment Means Supporting the Derivation of Detailed Success Criteria.

The relation of the Subtasks/Steps and the referenced sections in IAEA-TECDOC-1511 and AESJ standard [Ref-25.14] [Ref-25.21] is mapped in Table 25.4.2-1. Clearly some aspects of IAEA-TECDOC-1511 and AESJ standard cannot be implemented to the UK ABWR because of its pre-operational design status. The level of detail and extent to which those standards/guides are applied are commensurate with the level of UK ABWR design documentation.

To satisfy the success criteria in the Level 1 PSA model, each accident sequence must maintain a safe stable state for at least 24 hours.

25.4.2.1 Subtask 1: Definition of Overall and Detailed Success Criteria

Step 1: Definition of safety functions

Fundamental safety functions for the UK ABWR are defined in List of Safety Category and Class for UK ABWR as follows [Chapter 5 of PCSR Rev. C].

(1) Control of reactivity,

(2) Fuel cooling,

(3) Long term heat removal,

(4) Confinement/Containment of radioactive materials, and

(5) Others (support function to the above functions).

High level safety functions and key Systems, Structures and Components (SSCs) for each of the fundamental safety functions are listed in [Chapter 5 of PCSR Rev. C].

Meanwhile, the broad safety functions (frontline) considered for the UK ABWR IEAP Level 1 PSA are defined as follows.

(1) Reactivity Control, corresponding to (1) Control of reactivity in [Chapter 5 of PCSR Rev. C],

(2) Core Cooling, corresponding to (2) Fuel cooling in [Chapter 5 of PCSR Rev. C], and

(3) Containment Heat Removal, corresponding to (3) Long term heat removal in [Chapter 5 of PCSR Rev. C].

In addition to the above three functions, an additional three safety functions (frontline) have been defined for the success criteria derivation.

(4) Reactor Coolant Pressure Boundary Protection, corresponding to (4) Confinement/Containment of radioactive materials in [Chapter 5 of PCSR Rev. C],

(5) Vapour Suppression corresponding to (4) Confinement/Containment of radioactive materials in [Chapter 5 of PCSR Rev. C], and

(6) Containment Isolation, corresponding to (4) Confinement/Containment of radioactive materials in [Chapter 5 of PCSR Rev. C].

Step 2: Definition of overall success and acceptance criteria

Definition of core damage

In the UK ABWR PSA, "core damage" is defined as the condition that either or both of the following criteria have occurred.

- The peak cladding temperature (PCT) is 1,200 °C or greater.
- Total thickness of oxidation of the fuel cladding is 15 percent or greater of the total cladding thickness before oxidation.

Reactivity control

Since the success criteria for reactivity control have to be determined in the fault tree and event tree analyses for the PSA, the acceptance criterion is broadly defined as: "to achieve subcriticality and maintain the reactor in a subcritical state (combined with adequate core cooing and/or containment heat removal functions if appropriate) so that containment failure and subsequent core damage are avoided."

Core cooling

Failure of the "core cooling" function would cause core uncovery and lead to core damage. The acceptance criteria are identical to those for defining core damage.

- To maintain the peak cladding temperature (PCT) below 1,200 °C, and
- To keep total oxidation of the fuel cladding from exceeding 15 percent of the total cladding thickness before oxidation.

Containment heat removal

The acceptance criteria for preventing containment failure are defined in [Ref-25.49].

- To maintain the containment pressure below 620 kPa [gauge], and
- To maintain the containment temperature below 200 °C.

Reactor coolant pressure boundary protection

Failure to open the necessary number of SRVs in postulated pressurisation events, or failure to trip the necessary number of RIPs given failure of the RPS at transients, could result in loss of the RCPB, i.e. a consequential Large LOCA event. A pressure of 120 percent of the RCPB design pressure is chosen as the acceptance criterion.

Failure to reclose SRV(s) results in a unique accident sequence: Stuck Open Relief Valve (SORV). The acceptance criterion to avoid this situation is to reclose all the SRVs, which is identical to the actual success criterion.

Vapour suppression

There are several potential failure modes of the Vapour Suppression System (VSS). The following failures are modelled as VSS failure:

- W/W to D/W vacuum breakers fail to open during LOCA,
- SRV tall pipe break,
- SRV tall pipe check valves fail to open, and
- SRV tall pipe check valves fail to re-close.

Possibility and acceptance criteria for each failure mode were assessed.

Containment isolation

Failure to isolate the line penetrating both the RPV and containment when a BOC occurs could result in a BOC initiating event. The acceptance criterion is that the inboard isolation valve or outboard isolation valve is closed and/or kept closed without internal/external leak.

Step 3: Definition of success criteria

The success criteria for IEAP Level 1 PSA of UK ABWR were defined for each of the above safety functions, based on the supporting analyses performed in Subtask 2.

25.4.2.2 Subtask 2: Thermal Hydraulic Analyses and other Assessment Means Supporting the Derivation of Detailed Success Criteria

This section provides neutronics analyses and thermal hydraulic analyses supporting the success criteria defined in Subtask 1.

The analysis codes used are:

- The Lattice Analysis Code and 3-D Core Simulator [Chapter 24 of PCSR Rev. C][Ref-25.23],
- ODYN [Chapter 24 of PCSR Rev. C][Ref-25.23],
- TASC [Chapter 24 of PCSR Rev. C][Ref-25.23],

- SAFER [Chapter 24 of PCSR Rev. C][Ref-25.23],
- TRACG [Chapter 24 of PCSR Rev. C][Ref-25.23], and
- MAAP [Chapter 26 of PCSR Rev. C][Ref-25.22].

Step 1: Reactivity control

The reactivity control success criteria for the UK ABWR are supported by the core neutronics analyses using the same code(s) as was used for the UK ABWR core design [Chapter 6 of PCSR Rev. C] and Rod Withdrawal Events [Chapter 24 of PCSR Rev. C], and the thermal hydraulic analyses using the same code(s) as UK ABWR DBA [Chapter 24 of PCSR Rev. C][Ref-25.23].

Step 2: Core cooling

The core cooling success criteria for the UK ABWR are supported by the thermal hydraulic analyses using the same code(s) as UK ABWR DBA [Chapter 24 of PCSR Rev. C][Ref-25.23].

Step 3: Containment heat removal

The containment heat removal success criteria for the UK ABWR are supported by the thermal hydraulic analyses using the code for the UK ABWR severe accident analyses [Chapter 26 of PCSR Rev. C][Ref-25.49].

Step 4: Reactor coolant pressure boundary protection

The RCPB protection success criteria for the UK ABWR are supported by the thermal hydraulic analyses using the same code(s) as UK ABWR DBA [Chapter 24 of PCSR Rev. C][Ref-25.23].

Step 5: Vapour suppression function

Depending on the accident sequences and credited SSCs, the success criteria for the vapour suppression function were taken from the relevant DBA [Chapter 24 of PCSR Rev. C][Ref-25.23] or the failure conditions were incorporated into the thermal hydraulic analysis for the containment heat removal in Step 3.

Step 6: Containment isolation function

Supporting analyses are not required for this function because the success criterion is obvious, i.e. closing isolation valves.

Table 25.4.2-1 Relation of Subtasks/Steps and Referenced Sections of Guides/Standards

Subtask/Step	IAEA-TECDOC-1511 [Ref-25.14]	AESJ Level 1 PSA standard [Ref-25.21]
Subtask 1 Step 1 Step 2 Step 3	SC-A SC-A04 SC-A01,02 SC-A03	Section 6.1 Section 6.1.2, Explanation 11 Section 6.1.1, 6.1.4 Section 6.1.2, Explanation 11
Subtask 2 Step 1 Step 2 Step 3	SC-B	Section 6.1.5

25.4.3 Event Sequence Analysis

25.4.3.1 Subtask 3: Selection of a Method and Provision of Related Tools for Accident Sequences Modelling

This subtask follows the task AS-A01 in IAEA-TECDOC-1511 [Ref-25.14].

The method chosen for Accident Sequence Analysis provides for the possibility to explicitly model the appropriate combinations of system responses and operator actions that affect the key safety functions for each modelled initiating event /IE group and provides a framework to support sequence quantification. The method supports graphical representation of the accident sequence logic (e.g. 'event tree structure').

Event trees for the UK ABWR IEAP Level 1 PSA are developed using CAFTA (Computer Aided Fault Tree Analysis) code. CAFTA was developed by EPRI. This code is widely applied for PSA for existing plants in the US and for GDA in UK. The version of CAFTA used is 6.0b [Ref-25.32].

25.4.3.2 Subtask 4: Definition of Success and Non-Success End States and Key Safety Functions

This subtask follows the task AS-B in IAEA-TECDOC-1511 [Ref-25.14].

For each initiating event group the key safety functions that are necessary to reach a success end state are identified. Success and non-success end states are clearly defined.

Definition of Success End State

In the UK ABWR PSA, a success end state is achieved by "successful" operation of reactivity control, core cooling, and containment heat removal. The specific "success" of each function is discussed in Subtask 1.

Other safety functions, e.g. Reactor coolant pressure boundary protection, Vapour suppression, and Containment isolation, are not essential to achieve a success end state. Success or failure of these safety functions may impact the definition of non-success end states.

Since the containment heat removal function is examined at the end of the Level 1 PSA event trees, the "success end state" is one of the following.

(a) Success of the Power Conversion Systems (PCS) after success of the reactivity control and core cooling functions,

(b) Success of RHR (Suppression Pool (S/P) cooling mode, D/W spray mode or LPFL mode through the RHR heat exchangers) after success of the reactivity control and core cooling functions, or

(c) Success of the containment venting system after success of the reactivity control and core cooling functions.

For the containment venting end state, a claim may be made on the Containment Overpressure Protection System (COPS) according to the success of the core cooling system.

In addition to the above end states, successful core cooling even after containment failure or bypass is treated as a success end state in Level 1 PSA.

(d) Success of continued RPV injection by the systems outside the reactor building, i.e. feedwater system, condensate system or FLSS after containment failure, or

(e) Success of RPV injection by the systems outside the reactor building, i.e. condensate system or FLSS, after containment failure.

Definition of Non-Success End State

Failure to achieve any of the five success end states described above results in a non-success end state, i.e. core damage in Level 1 PSA. Accident Classes are used to characterise the non-success end states.

The identifiers used to characterise the Accident Classes are summarised in Table 25.4.3-1. Brief descriptions of the Accident Classes are summarised in Table 25.4.3-2.

Plant Damage States (PDSs) for containment event tree development are defined in Level 2 PSA based on the Accident Classes defined in Level 1 PSA (see Section 25.5).

Assessment of Success End States

The offsite consequence from the Level 1 PSA success end states are assessed against Hitachi-GE's NSEDP targets [Ref-25.58] (equivalent to SAP Targets 7 and 8) in Section 25.13. As the input to this assessment, the success end states are categorised into 10 groups.

- R : Success sequence by RHR (containment intact)
- C : Success sequence by containment venting (containment intact)
- CP : Success sequence with containment overpressure failure (with success of VSS)
- VP : Success sequence with containment failure (with VSS failure)
- B : Success sequence with containment bypass
- R* : Success sequence by RHR (containment intact) with perforated fuel
- C* : Success sequence by containment venting (containment intact) with perforated fuel
- CP* : Success sequence with containment overpressure failure (with success of VSS) with perforated fuel
- VP* : Success sequence with containment failure (with VSS failure) with perforated fuel
- B* : Success sequence with containment bypass with perforated fuel

All the success sequences are assigned to one of these groups.

25.4.3.3 Subtask 5: Accident Sequences Progression Identification and Accident Sequence Models Development

This subtask is the main part of event tree development. The role of an event tree is the graphical representation of the progression of an IE group, taking account of the identified success criteria (discussed in Subtasks 1 and 2) and assigning designated sequence end points (discussed in Subtask 4).

A simple example of what an event tree looks like is described below.

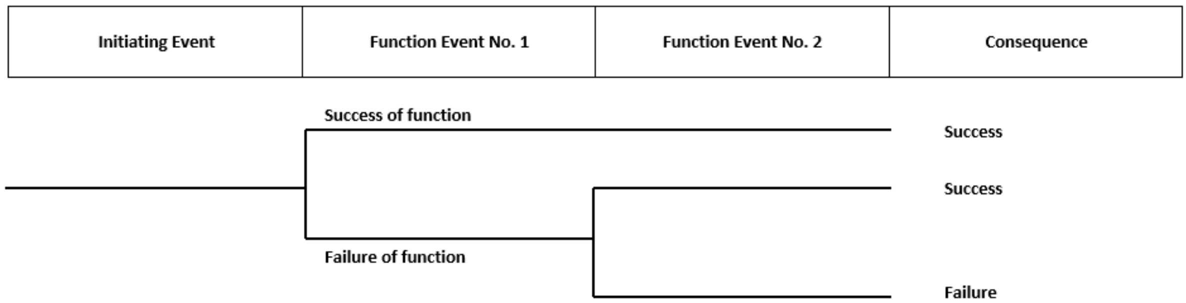

Figure 25.4.3-1 and Figure 25.4.3-2 present two examples of event trees modelled in the IEAP Level 1 PSA.

A brief description of the treatment of dependency, the use of consequential initiators, and the use of flags in the event trees is provided below.

Dependency treatment

Specific dependencies explicitly addressed in event tree development are:

- Systems directly or environmentally impacted by location specific LOCAs/BOCs/ISLOCAs are removed from event tree headings.
- Treatment of an end state with containment failure is distinguished by the success of core cooling systems.
- The manual shutdown event tree is used for SSIEs and tech spec initiators. Dependencies from the initiators are automatically captured by the integration of initiator fault trees and mitigating system fault trees.
- RCIC is bypassed given SORV, VSS failure or consequential LOCA.
- According to the thermal-hydraulic analyses, the RPV would be depressurised to the RCIC operable limit (1 MPa) within 1 hour given SORV and most of the LOCAs (except RPV bottom LOCAs). Whilst one hour operation of RCIC would delay the decrease in the water level, this does not have significant benefit to the PSA because subsequent water injection is required regardless of RCIC operation.
- Due to the potential impact of the SRV tailpipe failure (VSS failure) on RCIC (e.g. steam exhaust line), RCIC is not credited.

- HPCF is bypassed given LOOP with loss of EDGs.

- The power conversion system branch is bypassed given SORV, VSS failure, consequential LOCA, or failures of either the feedwater system or the condensate system.

- Degradation of RCIC operability is taken into account given a loss of Class 1 AC power or a loss of RHR.

NOTE: ECCS suction strainer blockage (including CCF) is explicitly modelled in the system fault trees regardless of the initiating event and accident sequence.

Consequential initiators

Similar plant PSAs has been reviewed to identify the consequential initiators. The following consequential initiators are considered in the event tree analysis.

- Conditional LOOP: Heading “C-LOOP”, “C-LOOP-LOCA”, link trees prepared for non-LOCA events

- Consequential LOCA: Heading “C-LOCA”, “SRV-O”

- Stuck Open Relief Valve: Heading “SORV”

Additional event trees are prepared for the following specific accident sequences.

- Failure to insert control rods (ATWS)

- Loss of Class 1 DC power supply at LOOP IE groups

- Group of sequences challenging Heat Capacity Temperature Limit (HCTL)

Flag settings

In the UK ABWR PSA, various condition settings are utilised as “flag files” in CAFTA. A flag file is applied to each non-success end state (core damage scenario) of each event tree in order to control the available SSCs, initiation signals, operator actions and so on.

A brief description of each of the event trees developed for the IEAP Level 1 PSA is provided in the reminder of this subsection below.

<u>Large LOCA Within Containment Event Trees (A, A-FWA)</u>

Large LOCA is defined as a break that depressurises the reactor to the point where the low pressure systems can automatically inject sufficient core cooling to prevent core damage.

The location specific LOCA groups are defined in the IE analysis task (see section 25.4.1) based on the design drawings and thermal-hydraulics analyses.

Medium LOCA Within Containment Event Trees (S1-BDL, S1-HPCFB, S1-HPCFC, S1-RHRB, S1-RHRC)

The Medium LOCA differs from Large LOCA in that the initial depressurisation rate is sufficiently slow to require SRV actuation (either ADS or RDCF) for rapid depressurisation in order to prevent core damage by low pressure injection. RCIC has insufficient capacity to maintain water level even for liquid LOCA in RPV bottom. The Medium LOCA differs from Small LOCA in that the decrease in the water level is too fast to credit the manual initiation of any injection system.

The location specific LOCA groups are defined in the IE analysis task (see section 25.4.1) based on the design drawings and thermal-hydraulics analyses.

Small LOCA Within Containment Event Trees (S2, S2-RHRB, S2-RHRC, S2-SLC, S2-RWLA, S2-RWLB, S2-RWLC, S2-RWLD)

In the Small LOCA, the initial depressurisation rate is sufficiently slow to require SRV actuation (either ADS or RDCF) for rapid depressurisation in order to prevent core damage by low pressure injection, which is the same as the Medium LOCA. For the steam LOCA (including initially liquid LOCAs above Top of Active Fuel), RCIC does not complete the mission time because of the reactor depressurisation. Small LOCA differs from Medium LOCA in that the decrease in the water level is slow enough to credit the manual initiation of HPCF.

The location specific LOCA groups are defined in the IE analysis task based on the design drawings and thermal-hydraulics analyses.

BOC Event Trees (S3-BMSFB, S3-BFWA, S3-BRCIC, S3-BCUW, S3-BSAM)

BOC is defined as a break outside containment of a pipe that is connected to the RPV and penetrates the containment, followed by automatic isolation failure (by motor operated valves, air operated valves or check valves). BOC differs from LOCA within containment in that

- An injection system taking suction from S/P cannot complete its mission time due to loss of water source / NPSH.
- Environmental impact from BOC should be considered (determined in IE analysis task (see section 25.4.1)).

The location specific LOCA groups are defined in the IE analysis task (see section 25.4.1) UK ABWR. The Large/Medium/Small categories are determined based on the same criteria (threshold break area) as for LOCAs within containment.

- BOC – Main steam line –: Large LOCA
- BOC – FW-B line –: Large LOCA
- BOC – FW-A line –: Large LOCA
- BOC – RCIC steam line –: Medium LOCA
- BOC – CUW line –: Large LOCA (with credit of FLSS)

- BOC – SAM line –: Small liquid LOCA

ISLOCA Event Trees (S3-IHLB, S3-IHLC, S3-IRA, S3-IRB, S3-IRC)

NUREG/CR-5124 [Ref-25.40] defines ISLOCA as LOCA in which the Reactor Coolant System pressure boundary (isolation valve, piping wall, etc.) interfacing with a supporting system of lower design pressure is breached.

In the PSA Accident Sequence Modelling, ISLOCA has the same features as BOC.

- An injection system taking suction from S/P cannot complete its mission time due to loss of water source / NPSH.

- Environmental impact from ISLOCA should be considered.

The location specific LOCA groups are defined in the IE analysis task (see section 25.4.1 UK ABWR. The Large/Medium/Small categories are determined based on the same criteria (threshold break area) as for LOCAs within containment, which are derived by thermal-hydraulic analysis.

- ISLOCA –HPCF-B injection line–: Medium LOCA

- ISLOCA –RHR-B (LPFL-B) injection line–: Medium LOCA

- ISLOCA –HPCF-C injection line–: Medium LOCA

- ISLOCA – RHR-C (LPFL-C) injection line–: Medium LOCA

- ISLOCA –RHR-A suction line–: Large LOCA

- ISLOCA –RHR-B suction line–: Large LOCA

- ISLOCA –RHR-C suction line–: Large LOCA

NOTE: The largest area given guillotine break is limited by the narrowest flow path, e.g. nozzle, sparger, flow limiter.

Excessive LOCA Event Tree (S4)

Excessive LOCA is defined as the LOCA in which core damage is not avoided. In the PSA, this is represented by RPV rupture.

Transient Event Trees

General Transient Event Tree (TG)

The general transient group in the UK ABWR PSA is defined as a transient with intact pressure boundary, available offsite power, non-isolation of the main steam line, available feedwater system and available main condenser, which means a transient with no dependent failure.

This event tree is also used for another transient (PCV pressurization by leakage of nitrogen gas) because this transient has similar accident progression and success criteria to those of the general transient.

Manual Shutdown and Support System Initiator Event Tree (MS)

IE analysis task identified various initiating events leading to manual shutdown.

- Manual shutdown (including Tech Spec Initiators)
- Loss of Class 1 AC
- Loss of Class 3 AC
- Loss of Class 1 DC
- Loss of Class 2 DC
- Loss of RCW/RSW
- Loss of R/BEEE/Z HVAC
- Loss of TBNEE/Z HVAC
- Loss of T/B HVAC
- Loss of Hx/B-N HVAC
- Loss of CBEEE/Z (A) HVAC
- Loss of CBEEE/Z (B) HVAC
- Loss of CBEEE/Z (C) HVAC
- Loss of MCR HVAC
- Loss of HNCW
- CCF of Class 1 controller
- CCF of Class 3 controller
- Loss of Instrument or Control Air System

Since these initiators do not immediately disturb plant normal operation, the plant is manually shutdown (if the initial fault(s) is not corrected within the duration allowed by tech-spec).

NOTE: The initiators are modelled as fault trees. Dependent failures from the initiators on the mitigation systems are automatically captured by the master fault tree approach. All the initiator fault trees are connected to the MS event tree under an OR gate.

ATWS Link Trees (TG-A, TM-A, TF-A, TI-A, TE1-A, TE2-A, TE3-A, TE4-A)

These event trees analyse the accident sequences after the reactivity control top event F-CR or F-CRE in the transient event trees.

Loss of Class 1 DC Link Trees (TE1-DC, TE2-DC, TE3-DC, TE4-DC)

These event trees analyse the accident sequences after failure of the Class 1 DC top event DC1 in the LOOP event trees.

Conditional LOOP Link Trees (TG-LOOP1, 2, 3, 4, TM-LOOP1, 2, 3, 4, TF-LOOP1, 2, 3, 4, TI-LOOP1, 2, 3, 4)

These event trees analyse the accident sequences after Conditional LOOP (C-LOOP) in the transient event trees.

HCTL Link Trees (TG-HCTL, TM-HTCL, TF-HCTL, TE1-HCTL, TE2-HCTL, TE3-HCTL, TE4-HCTL, MS-HCTL)

These event trees analyse the accident sequences after success of RCIC or HPCF and failure of RHR S/P cooling mode in the transient event trees.

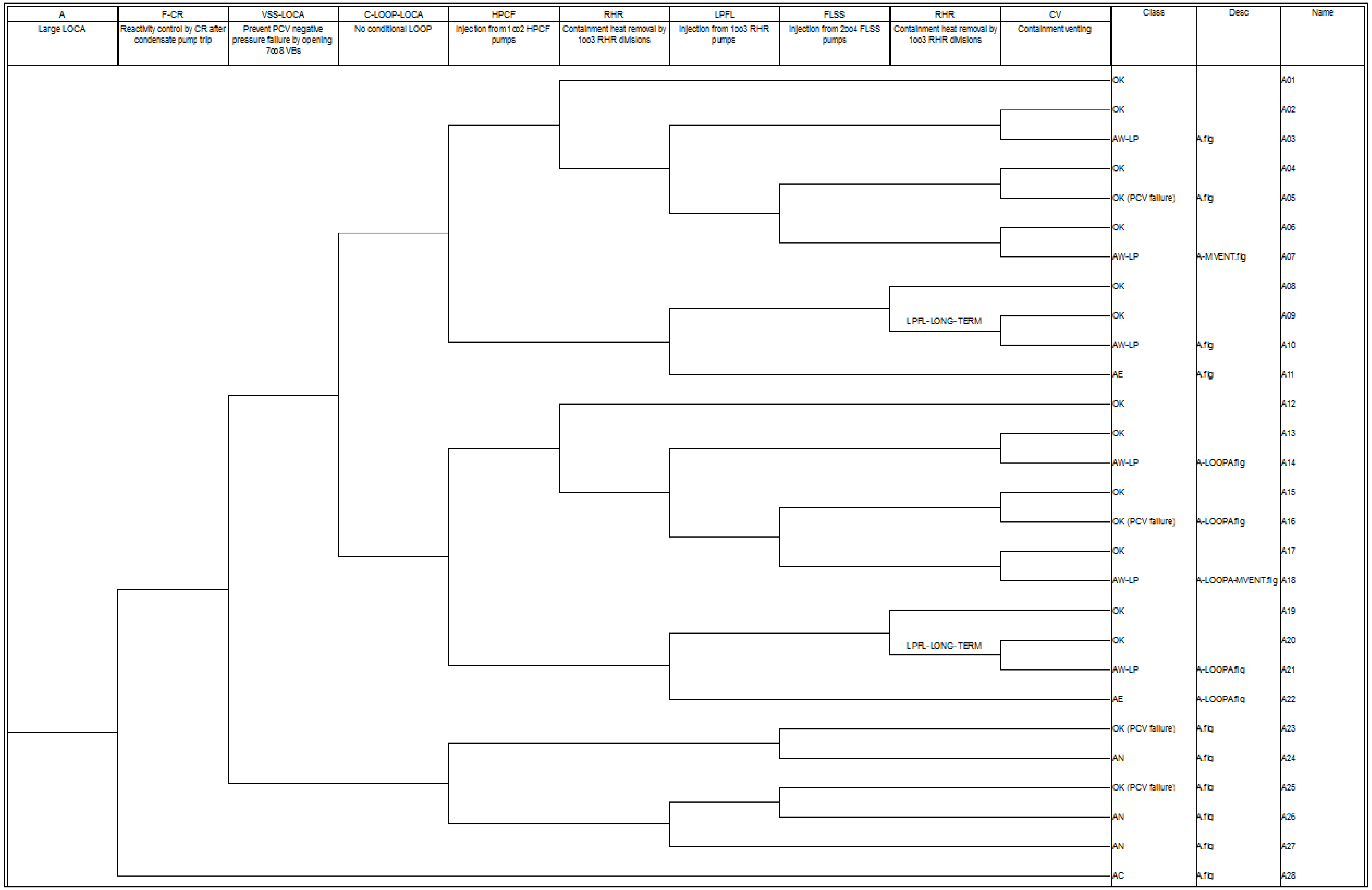

Figure 25.4.3-1 Event Tree A for Large LOCA (MSL, SRV Inlets, FW-B, RHR-A, B, C suction, CUW mid-vessel suction) within Containment

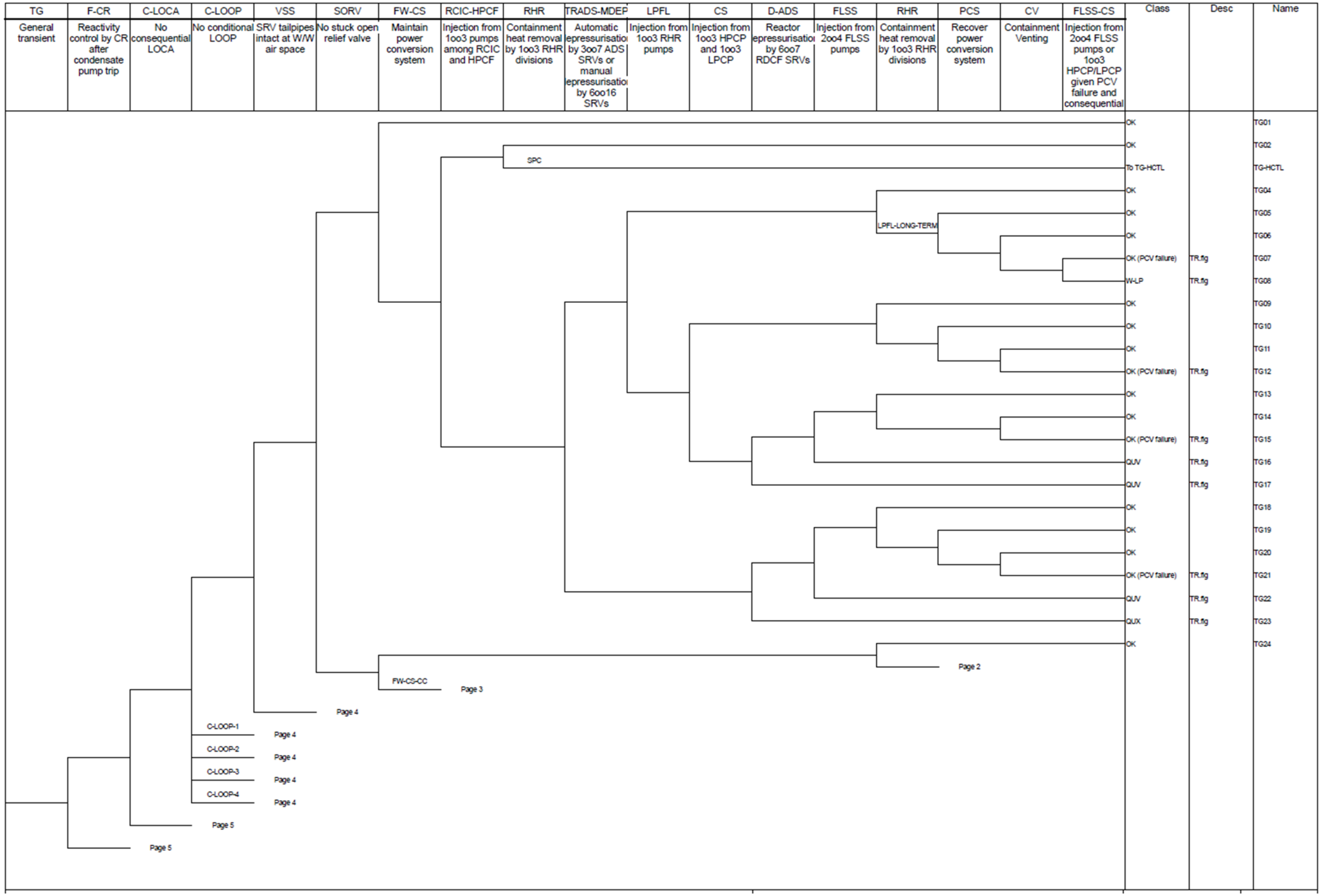

Figure 25.4.3-2 Event Tree TG for General Transient and PCV Pressurisation by Leakage of Nitrogen Gas

Table 25.4.3-1 Identifiers Used in Accident Classes

ID	Description
A	LOCA inside containment
B	Loss of AC power for Class 1 and Class 2 safety systems
C	PCV failure in short term and prior to core damage due to failure of reactivity control
C-HP	Core damage at high pressure in short term and prior to PCV failure due to failure of reactivity control and core cooling
C-LP	Core damage at low pressure in short term and prior to PCV failure due to failure of reactivity control and core cooling
D	Loss of DC power for Class 1 and Class 2 safety systems
E	Failure of high pressure and low pressure injection at LOCA
N	Vapour Suppression System Failure[1]
P	Failure to reclose SRV under loss of Class 1 AC power
Q	Failure of feedwater system
S12	Medium and small LOCAs inside containment (liquid LOCA at RPV bottom)
S3	ISLOCA or Break Outside Containment (BOC)
S4	Excess LOCA[2]
T	Transient[3]
U	Failure of high pressure ECCS
V	Failure of low pressure injection
W	Failure of heat removal from PCV followed by core damage at high pressure
W-LP	Failure of heat removal from PCV followed by core damage at low pressure
X	Failure of reactor depressurisation

[1]This Identifier has two meanings: (1) Failure to open W/W to D/W vacuum breakers in LOCA events followed by RPV injection from external water source, (2) SRV tailpipe break at W/W in non-LOCA events

[2]Directly leading to core damage (containment integrity assessed in Level 2 PSA)

[3]Includes planned manual shutdown and SSIEs

Table 25.4.3-2 Descriptions of Accident Classes

Class	Description
S4	Excess LOCA inside containment, directly resulting in low pressure core damage in short term
AE	LOCA inside containment (including consequential LOCAs) with failure of RPV injection, resulting in low pressure core damage in short term
S12UX	Medium LOCA or small LOCA at RPV bottom inside containment with failure of high pressure injection and depressurisation, resulting in high pressure core damage in short term
S3E	ISLOCA or BOC with failure of RPV injection, resulting in low pressure core damage in short term with containment bypass NOTE: ISLOCA/BOC with failure of reactivity control is also grouped as S3E due to similar end state expected, i.e. low pressure core damage in short term with containment bypass
S3UX	BOC (small liquid LOCA) with failure of high pressure injection and depressurization, resulting in high pressure core damage in short term with containment bypass NOTE: BOC (small liquid LOCA) with failure of reactivity control is also grouped as S3UX due to similar end state expected, i.e. high pressure core damage in short term with containment bypass
TQUV	Non-LOCA event with failure of high pressure injection and low pressure injection, resulting in low pressure core damage in short term
TQUX	Non-LOCA event with failure of high pressure injection and depressurization, resulting in high pressure core damage in short term
TB	Loss of Class 1 and Class 2 AC power, resulting in high pressure core damage after 8 hours operation of RCIC
TBU	Loss of Class 1 and Class 2 AC power with failure of RCIC, resulting in high pressure core damage in short term
TBP	Loss of Class 1 and Class 2 AC power with failure to reclose SRV (RCIC inoperable), resulting in low pressure core damage in short term
TBD	Loss of Class 1 and Class 2 AC and DC power, resulting in high pressure core damage in short term
TW	Non-LOCA event (RPV pressure kept high) with failure of containment heat removal followed by failure of HPCF, resulting in high pressure core damage in long term after containment failure
S12W	Medium LOCA or small LOCA at RPV bottom inside containment with failure of containment heat removal followed by failure of HPCF, resulting in high pressure core damage in long term after containment failure
TW-LP	Non-LOCA event (RPV depressurised) with failure of containment heat removal followed by failure of LPFL or HPCF, resulting in low pressure core damage in long term after containment failure
AW-LP	LOCA inside containment (including consequential LOCAs) with failure of containment heat removal followed by failure of LPFL or HPCF, resulting in low pressure core damage in long term after containment failure
AC	LOCA inside containment (including consequential LOCAs) with failure of reactivity control, resulting in containment failure in short term followed by core damage at low pressure
TC	Non-LOCA event with failure of reactivity control, resulting in containment failure in short term followed by core damage at high pressure
S12C	Medium LOCA or small LOCA at RPV bottom inside containment with failure of reactivity control, resulting in containment failure in short term followed by core damage at high pressure
TC-HP	Non-LOCA event with failure of control rod insertion and core cooling, resulting in high pressure core damage in short term
TC-LP	Non-LOCA event with failure of control rod insertion and core cooling, resulting in low pressure core damage in short term
TNQUV	Non-LOCA event with SRV tailpipe break at W/W airspace, failure of high pressure injection and low pressure injection, resulting in low pressure core damage in short term with S/P bypass (containment failure before core damage due to RPV depressurisation by SRVs)
TNQUX	Non-LOCA event with SRV tailpipe break at W/W airspace, failure of high pressure injection and depressurization, resulting in high pressure core damage in short term with S/P bypass
TBPN	Loss of Class 1 and Class 2 AC power with SRV tailpipe break at W/W airspace, resulting in high pressure core damage in short term with S/P bypass
TBDN	Loss of Class 1 and Class 2 AC/DC power with SRV tailpipe break at W/W airspace, resulting in high pressure core damage in short term with S/P bypass
TCN	Non-LOCA event with failure of control rod insertion and SRV tailpipe break at W/W airspace, resulting in containment failure in short term followed by core damage at high pressure with S/P bypass Non-LOCA event with success of control rod insertion, SRV tailpipe break at W/W airspace and failure of containment heat removal , resulting in containment failure in short term followed by core damage at high pressure with S/P bypass
AN	LOCA inside containment (including consequential LOCAs) with failure of W/W to D/W vacuum breakers to open and failure of RPV injection, resulting in containment failure in short term followed by low pressure core damage

25.4.4 System Analysis

System analysis assesses the reliability of those systems designed to meet the safety requirements identified in Section 25.4.3. This generally involves the production of fault trees (see Section 25.4.4.3). All system boundaries must be established prior to modelling (see Section 25.4.4.1) and modelling principles should be determined (see Section 25.4.4.2). The assessment requires the identification of failures that could result in the loss of the system's function. The probability of each type of failure occurring is then calculated and the failures (i.e. cutsets) can be ranked by decreasing order of probability. This will reveal any potential weaknesses in the plant. Reliability values assigned to the identified failures (i.e. failure rates and probabilities) must be based on data which are representative of the plant operating experience (see Section 25.4.6). Fault tree development includes component failures, human failure events (see Section 25.4.5), and common cause failures (see Section 25.4.7).

25.4.4.1 Systems information required

System characteristics including system boundaries have been defined for all systems, including support systems, needed for performing the mitigation functions identified in the accident sequence analysis (see Section 25.4.3). Plant information sources were reviewed in order to:

- Define the system function during normal and accident conditions,
- Establish system boundaries,
- Identify interfaces with other systems,
- Identify instrumentation and control requirements including the operator interface,
- Identify testing and maintenance requirements ,
- Identify operating limitations such as those imposed by technical specifications, and
- Identify procedures for the operation of the system.

The following design information was used for performing the tasks described above:

- P&ID (Piping &Instrumentation Diagram),
- SDD (System Design Description),
- IBD (Internal Block Diagram),
- One-line Diagram,
- ECWD (Elementary Control Wiring Diagram),
- Periodic Test Procedure,
- Operational Procedure, and
- General Arrangement Drawings.

Some of the detailed design information required was not available in the GDA phase. Where this was the case, assumptions have been made based on surrogate ABWR in order to obtain the necessary information for the GDA PSA. Assumptions applied in the PSA development have been reviewed in order to guide future UK ABWR design work.

25.4.4.2 System modelling

This section identifies the mitigation systems which have been modelled in the IEAP Level 1 PSA, and describes the methodology applied in the modelling of each of those systems.

The following systems have been modelled in the PSA:

(1) Frontline systems necessary for preventing core damage,

(2) Support systems for each component in the frontline system, and

(3) Systems affecting the systems identified in (1) and (2).

The specific systems modelled in the IEAP Level 1 PSA are shown in Table 25.4.4-1, grouped into each of the associated functions. Table 25.4.4-1 also presents the system code assigned to each system. This code is used in the system modelling to allow unambiguous identification of all systems and components modelled.

The system modelling was developed in line with the ONR TAG, IAEA-TECDOC-1511 [Ref-25.14] and ASME/ANS PRA Standard [Ref-25.13].

Systematic FMEAs were performed to identify the credible failure modes including the Test & Maintenance events.

25.4.4.3 Fault tree development

The purpose of the system analysis is to model the system failures that were identified in the event sequence analysis (see Section 25.4.3) and identify the relevant failure modes contributing to the system failures through systematic FMEAs. The causes of the system failure are represented by the specific failure modes of individual components, taking into account the necessary function of the system. Quantification of the model then creates minimal cutsets for the system or group of systems in each of the event tree sequences, which represent the minimum combinations of component failures which lead to the end state of each sequence. These minimal cutsets are then used to capture the sources of risk.

The system models consist of fault trees whose top events are failure of a system to perform its required function which was identified in the “Event Sequence Analysis”. The various ways in which the top event may occur are developed through the branches of the tree. The fault tree development terminates at basic events associated with potential failures of system trains or components. The system fault trees contain sufficient detail to ensure that all credible failure modes are considered and the resulting model accurately reflects the impact of each failure mode on system performance.

The objective of system analysis task is to create realistic system responses to accidents that could be integrated and used in the final sequence quantification. Other objectives include:

(1) Consideration of all credible failure modes by systematic FMEAs,

(2) Accurate modelling of each failure mode which has an impact on system performance,

(3) Explicit inclusion of all support system failure modes in the front-line system fault trees at the appropriate points,

(4) Creation of the adequate linkage among the fault tree models, and

(5) Creation of fault tree models whose solutions (i.e. cutsets) could be easily understood.

The system analysis consists of following three subtasks:

- Subtask 1: Development of the system description,
- Subtask 2: Failure mode analysis, and
- Subtask 3: Construction of system fault trees.

Subtask 1: Development of the system description

The analysis for each system begins by defining the system's functions, establishing system level success criteria (in conjunction with the event tree development), and defining the system boundaries. Other relevant system information is also provided that could provide background about system operation and enhance the model credibility and realism.

This subtask establishes the basis for the subsequent system modelling by obtaining all the necessary information about the system. This is accomplished through a systematic process of information collecting and documentation for each system, which includes the following steps:

- Step 1: Information collecting of system function,
- Step 2: Information collecting of system boundary and configuration,
- Step 3: Defining system success criteria,
- Step 4: Information collecting of system interfaces and dependencies,
- Step 5: Information collecting of test and maintenance, and
- Step 6: Information collecting of operational procedure for each system.

Subtask 2: Failure mode analysis

The main objective of the "failure mode analysis" subtask is to determine all the causes of system failures, which then forms the basis of fault tree development. This was accomplished through a systematic process of failure cause identification, which included the following steps:

- Step 1: Identification of failure states of component
- Step 2: Identification of failure causes of component
  - Mechanical failures
  - Failure of signal (failure of initiation signal / spurious signal)

  - Failure of support systems for the component
  - Human errors
  - Test and maintenance unavailability

- Step 3: Developing the way of modelling basic events
- Step 4: Modelling dependent failures
- Step 5: Summarising references and assumptions

A basic event was then created to represent each failure mode of each of the identified components. The basic event probability of failure was determined based on assigned parameters such as failure rate and test interval. Common cause failure was also considered as a cause of failure of groups of similar components.

Subtask 3: Construction of system fault trees

The main objective of the subtask "construction of fault trees" is to develop system fault trees for each of the mitigating systems identified in the event sequence analysis (see Section 25.4.3). The basic structure of the fault trees is determined by the failure mode analysis described above.

In general, gates in the system fault trees have been identified using nomenclature which follows a consistent naming scheme. Similarly, individual components and combinations of components (basic events) follow a consistent naming scheme. This ensures that combinations of events which lead to undesired plant state are readily understandable when the PSA model is quantified:

Subtask 3, "construction of system fault trees", included the following steps.

- Step 1: Developed the basic structure of each identified system fault tree,
- Step 2: Coded the fault tree based on consistent naming conventions, and
- Step 3: Took account of the dependency of each system failure on the initiating event.

Table 25.4.4-1 Systems Modelled in Internal Events at Power Level 1 PSA

<table>
<tr><th>Function</th><th colspan="2">System</th><th>System Code</th></tr>
<tr><td rowspan="5">Reactivity Control Function</td><td colspan="2">Reactor Protection System (RPS)</td><td>C71</td></tr>
<tr><td colspan="2">Control Rod Drive System (CRD)</td><td>C12</td></tr>
<tr><td rowspan="2">Anticipated Transients Without Scram System (ATWS)</td><td>Alternative Rod Insertion (ARI)</td><td rowspan="2">C73</td></tr>
<tr><td>Recirculation Pump Trip (RPT)</td></tr>
<tr><td colspan="2">Standby Liquid Control System (SLC)</td><td>C41</td></tr>
<tr><td rowspan="6">Core Cooling Function</td><td colspan="2">Reactor Core Isolation Cooling System (RCIC)</td><td>E51</td></tr>
<tr><td colspan="2">High Pressure Core Flooder System (HPCF)</td><td>E22</td></tr>
<tr><td colspan="2">RHR-Low Pressure Core Flooder System (LPFL)</td><td>E11</td></tr>
<tr><td colspan="2">Automatic Depressurisation System (ADS)</td><td>B21</td></tr>
<tr><td colspan="2">Flooder System of Specific Safety Facility (FLSS)</td><td>E71</td></tr>
<tr><td colspan="2">Flooder System of Reactor Building (FLSR)</td><td>E72</td></tr>
<tr><td rowspan="4">Heat Removal Function</td><td rowspan="2">RHR</td><td>Suppression Pool Cooling mode</td><td>E11</td></tr>
<tr><td>Drywell Spray mode</td><td>E11</td></tr>
<tr><td colspan="2">Filtered Containment Venting System (FCVS)</td><td>T61</td></tr>
<tr><td colspan="2">Atmospheric Control system (AC)</td><td>T31</td></tr>
<tr><td rowspan="13">Support System</td><td rowspan="7">Electrical Power Distribution system (R10)</td><td>AC Power Supply</td><td>-</td></tr>
<tr><td>- Metal-Clad Switchgear (M/C)</td><td>R22</td></tr>
<tr><td>- Power Centre (P/C)</td><td>R23</td></tr>
<tr><td>- Motor Control Centre (MCC)</td><td>R24</td></tr>
<tr><td>Emergency Diesel Generator System (EDG)</td><td>R43</td></tr>
<tr><td>Back-up Building Generator System (BBG)</td><td>R44</td></tr>
<tr><td>DC Power Supply</td><td>R42</td></tr>
<tr><td colspan="2">Reactor Building Cooling Water System (RCW)</td><td>P21</td></tr>
<tr><td colspan="2">Reactor Building Service Water System (RSW)</td><td>P41</td></tr>
<tr><td colspan="2">Heating Ventilating and Air Conditioning System (HVAC)</td><td>U41</td></tr>
<tr><td colspan="2">HVAC Normal Cooling Water System (HNCW)</td><td>P24</td></tr>
<tr><td colspan="2">HVAC Emergency Cooling Water System (HECW)</td><td>P25</td></tr>
<tr><td colspan="2">I & C (including digital system)</td><td>A32</td></tr>
<tr><td rowspan="3">Core Cooling/ Heat Removal Function (Others)</td><td colspan="2">Feedwater System</td><td>B21,N21</td></tr>
<tr><td colspan="2">Condensate System</td><td>N21</td></tr>
<tr><td colspan="2">Power Conversion Systems (PCS)</td><td>(N21)</td></tr>
<tr><td rowspan="14">Support System (Others)</td><td colspan="2">Main Steam System (for PCV Isolation)</td><td>B21</td></tr>
<tr><td colspan="2">Reactor Water Clean-up System (for PCV Isolation)</td><td>G31</td></tr>
<tr><td colspan="2">Air Off Take System (AO)</td><td>N21</td></tr>
<tr><td colspan="2">Circulating Water System (CW)</td><td>N71</td></tr>
<tr><td colspan="2">Off-Gas System (OG)</td><td>N62</td></tr>
<tr><td colspan="2">Turbine Building Cooling Water System (TCW)</td><td>P22</td></tr>
<tr><td colspan="2">Turbine Building Service Water System (TSW)</td><td>P42</td></tr>
<tr><td colspan="2">Turbine Gland Steam system (TGS)</td><td>N33</td></tr>
<tr><td colspan="2">Instrument Air System (IA)</td><td>P52</td></tr>
<tr><td colspan="2">Station Service Air System (SA)</td><td>P51</td></tr>
<tr><td colspan="2">High Pressure Nitrogen Gas Supply System (HPIN)</td><td>P54</td></tr>
<tr><td colspan="2">Heating Steam System</td><td>P61</td></tr>
<tr><td colspan="2">Make-up Water Condensate System (MUWC)</td><td>P13</td></tr>
<tr><td colspan="2">Make-up Water Purified System (MUWP)</td><td>P11</td></tr>
</table>

25.4.5 Human Reliability Analysis

This section describes the strategy/methodology of the specific Human Reliability Analysis (HRA) performed for the IEAP Level 1 PSA. Relevant requirements of IAEA-TECDOC-1511 [Ref-25.14] are referenced.

[Chapter 27 of PCSR] and Human Reliability Analysis Report [Ref-25.44] describe the technical Human Factors work performed, including HRA.

HRA was carried out on the pre-initiator type human failure events (Type A HFEs) for the PSA as follows:

- Subtask 1 Identification of potential human failure events that result in loss of function of mitigation systems,
- Subtask 2 Qualitative HEA/HRA for pre-initiator HFEs, and
- Subtask 3 HEP quantification for pre-initiator HFEs.

HRA was carried out on the post-initiator type human failure events (Type C HFEs) for the PSA as follows:

- Subtask 4 Determination of post-initiator operator actions credited for PSA,
- Subtask 5 Qualitative HEA/HRA for post-initiator HFEs, and
- Subtask 6 HEP quantification for post-initiator HFEs.

Note that although the UK ABWR is an evolutionary design based on operational plant, it is only possible to conduct a preliminary high-level identification and analysis of potential Type A HFEs during GDA because the specific equipment maintenance regime for UK plant has not yet been developed. This means that detailed procedures that are required to provide a robust analysis of maintenance tasks are not yet available. This will be addressed in post GDA phase of PSA.

25.4.5.1 Subtask 1: Identification of potential human failure events that result in loss of function of mitigation systems

Potential pre-initiator HFEs that may result in loss of function of the mitigation systems credited in PSA were identified through the following process.

Step 1: Review of Operating Experience

Operating Experience (OE) is considered to provide the best resource of information on the types or categories of pre-initiator HFEs. The review of OE included both BWR experience, and BWR/ABWR PSA modelling, such as the US ABWR PSA.

Step 2: Identification of failure modes caused by human failure events

The process of identifying pre-accident human errors involves a review of:

- test procedures and activities for existing ABWR plants where a mechanical component is taken out of service for either corrective or preventive maintenance,

- instruments modelled in the PSA which require periodic surveillance testing and calibration,
- the plant design including P&IDs, electrical diagrams, and I&C drawings, and
- system fault trees.

Step 3: Identified HFEs are screened under consideration of PRA Standard Requirements for Pre-initiating Events

All HFEs in pre-initiating events were identified in Step 2 above. These HFEs were screened in accordance with the ASME/ANS PRA standard [Ref-25.13].

25.4.5.2 Subtask 2: Qualitative HEA/HRA for pre-initiator HFEs

These qualitative evaluations were made for generic HFE groups such as miscalibration of reactor water level instruments (single division and CCF of redundant divisions), restoration errors of manual valves (single valve and CCF of redundant valves), restoration errors of circuit breakers (single breaker and CCF of redundant breakers).

The critical restoration steps for each HFE group were identified based on experience from previous plant procedures and previous PSA analysis for similar plants. These steps involve restorations or other activities that if not properly performed, would render the component inoperable if not recovered.

Example 1 - Prior to performing maintenance on an HPCF pump, the protective tagging order typically involves isolating the pump by closing both the suction and discharge isolation valves, and racking out the pump breaker. Failure to return these components to their original position results in unavailability of the pump.

Example 2 - In the case of testing, failure to close the test line isolation valve at the end of the test may result in unacceptable degraded performance of the train/system.

Example 3 - During instrument calibrations, a technician could miscalibrate an instrument such as a trip unit or transmitter. In addition, typically in the case of a transmitter, the instrument isolation valve(s) could be left closed, which could result in erroneous operation of failure to detect a change in the operating conditions.

Once the critical components that are subject to a potential restoration error are determined, the possible error modes were evaluated (e.g., errors of omission and/or commission). Potential recovery mechanisms were evaluated to identify instances where an error by the original performer was recoverable. The recoveries include the following possible recovery steps:

- IV: Independent Verification is performed on the specific step(s),
- PM: Post-Maintenance testing is performed which would detect the error,
- PC: Post-Calibration testing is performed which would detect the error,
- SC: Shift check is performed using a checklist which would detect the error. A shift check may be either each operating shift, daily or weekly. Checks performed less often are not credited in the PSA, and

- AN: Annunciation would result if the error occurs.

25.4.5.3 Subtask 3: HEP quantification for pre-initiator HFEs

The generic pre-initiator HFEs defined in the previous subtask were quantified.

For individual (non-CCF) HFEs, Tables 20-5 through 20-26 of NUREG/CR-1278 [Ref-25.25] were used. The recovery factors assumed some level of dependence using the NUREG/CR-1278 Table 20-21 values [Ref-25.25].

For CCF HFEs (e.g. restoration errors of redundant components), in general, actions to restore components on multiple trains would not be performed closely in time, but would be performed using the same procedures, with check offs in the procedures. As a result, a low dependency was assessed for most restoration errors. The exception was for valves or other components on the same train, which would be restored closely in time, following the same procedure with check offs. In this case, a high dependence was assigned. These dependencies result in one of two values being assigned to the “dependent” restoration error of a similar or related component. The factors taken from Table 20-21 of NUREG/CR-1278 [Ref-25.25] for relevant level of dependency were multiplied to the individual non-CCF HFEs for calculating the CCF HFEs.

25.4.5.4 Subtask 4: Determination of Post-initiator Operator Actions Credited for PSA

A systematic process has been used to identify key human response actions that the operators may need to take in response to possible accident sequences and need to be modelled in the PSA. The post-initiator operator actions credited for PSA have been determined and implemented through the following process.

Step 1: Selection of the systems which deliver safety functions

The systems which deliver safety functions have been selected through the following process.

- Identification of post-initiator HFEs includes the following types of events:
  - Actions required to initiate (for those systems not automatically initiated), operate, control, isolate, or terminate those systems and components used in preventing or mitigating core damage as defined by the success criteria (e.g. operator initiates PCS).
- The following sources of plant information have been reviewed as part of the accident sequence analysis and systems analysis to identify the post-initiator actions which need to be modelled:
  - Emergency Operating Procedures (EOPs),
  - Abnormal Operating Procedures (AOPs) of existing ABWR plants,
  - Alarm response procedures of existing ABWR plants,
  - System operating procedures of existing ABWR plants, and
  - The design information.
- System Design Description (SDD) for individual systems

- Basis of Safety Cases (BSCs) for individual systems

  - Other procedures and design information.

Note that the selection of systems which deliver safety functions is a combined process with the event sequence analysis task and the system analysis task.

Actions to restore failed components such as restoring a feedwater system or RHR etc. are conservatively not credited because it is difficult to identify a failure mode of the failed system. This is conservative.

Step 2: Determination of operator actions

Through the identification process, post-initiator HFE "candidates" are identified and some of candidates are screened from the view point of feasibility e.g. procedure, environment and time available and/or expected risk reduction.

Step 3: Assignment of unique IDs to Human Failure Events

This is the shared process with the system analysis task. The unique IDs used for the PSA are also used in the HRA, i.e. qualitative HEA/HRA and HEPs quantification.

In addition, unique numbers are assigned to the identified HFEs.

Step 4: Linkage of post-initiator HFEs to event tree branches

Although the post-initiator HFEs are directly linked to the fault trees, explicit assignment of the HFEs to the event tree branches is important to understand the accident sequences where the operator actions are credited. Specifically, the unique numbers of the HFEs are illustrated in the PSA event trees for the purpose of familiarisation to the HF team and documentation.

Step 5: Representation of inter HFEs dependency

As is the nature of accident mitigation, multiple HFEs would appear within single accident sequence. In order to explicitly consider the dependency among cognitions for the needs of the same safety function, e.g. core cooling or long term heat removal, the following cognition failures are separately introduced and assigned to all the relevant branches and fault trees to which the related HFEs are assigned.

- Cognition error for manual operation of core cooling

- Cognition error for manual operation of long term heat removal

Dependency among cognitions for core cooling and long term heat removal is considered to be negligible due to the significantly different timing and objectives and is not considered further.

Step 6: Linkage of post-initiator HFEs to fault trees

This is the shared process with the system analysis task.

The cognition error for core cooling has been commonly linked to all the relevant fault trees in addition to the individual HFEs on core cooling. Also, the cognition error for long term heat removal has been commonly linked to all the relevant fault trees in addition to the individual HFEs on long term heat removal.

25.4.5.5 Subtask 5: Qualitative HEA/HRA for post-initiator HFEs

This task has been performed using a multidisciplinary team including the Human Factors team, members of the design sections (mainly system, C&I), operational specialists and members of the PSA team. Qualitative HEA/HRA has been performed [Chapter 27 of PCSR] [Ref-25.44].

In this Section, the necessary input from the PSA for the Qualitative HEA/HRA is defined.

- The accident sequences in which the operator actions are credited (use the graphical illustration on the event trees prepared in Subtask 4)
- Representative / bounding accident scenario for Qualitative HEA/HRA of each HFEs
  - Initiator
  - Systems attempted
  - Availability of systems in the scenario
  - Consequence from overall task failure
- Time available for each task based on the success criteria analysis and/or existing procedures
- Expected risk reduction by the analysed tasks / Expected risk increase if not credited

In addition to the Qualitative HEA/HRA, inter HFEs dependency is assessed to support the treatment in Step 5 of Subtask 4 [Ref-25.44].

25.4.5.6 Subtask 6: HEP quantification for post-initiator HFEs

Based on the qualitative HEA/HRA in subtask 5, the HEPs were quantified using the probabilities and the Performance Influencing Factors (PIFs) [Ref-25.25] [Ref-25.44] [Ref-25.45].

25.4.6 Data Analysis

Generic reliability data from the 2010 update to NUREG/CR-6928 [Ref-25.26], the 7th edition of the T-Book [Ref-25.27], WSRC-TR-93-262 [Ref-25.28], EH-33 [Ref-25.29] and IEEE std-500-1984 [Ref-25.30] have been used, with the last two references only being used for one component type each.

The NUREG/CR-6928 and T-Book databases are regularly updated, with the NUREG/CR-6928 study based on the large amount of nuclear industry operating experience collected in the US, including BWRs. The T-Book collates data from Nordic Nuclear power plants, where BWRs also operate. WSRC-TR-93-262 is a database developed by the Savannah River Site, which compiled data from a range of sources but is heavily populated with US commercial nuclear plant data and is used as one of the references for NUREG/CR-6928. The EH-33 study was performed for the US Department of Energy and presents data from a range of Incident Reporting systems, not necessarily from the nuclear industry. The IEEE guidance presents reliability data for electrical systems in nuclear stations but has not been updated or supported since the mid-1980s.

There is no publically available database of UK nuclear operating experience. There are also no operating BWRs in the UK at this time. Whilst there is data from Japanese BWRs, it has been found that the

Japanese data shows very high reliability when compared to other international data and would therefore not be appropriate for a UK plant. Therefore, the data sources listed above are the most appropriate to be using at this time.

In addition, failure rates could be higher for a POS during plant shutdown for shutdown PSA and SFP PSA if the historical data shows that component failures occur more often during that POS. However, at this stage in the design process there is no plant specific data available. Therefore, the failure rates applied in the shutdown PSA and the SFP PSA are the same failure rates applied in the at Power PSA (see sections 25.8 and 25.9 respectively).

25.4.7 Common Cause Failure Analysis

CCFs are a subset of dependent failures (i.e. those failures that defeat the redundancy or diversity that is employed to improve the availability of plant safety functions). CCFs, similar to other dependent failures, are addressed in the UK ABWR PSA by incorporating appropriate common cause basic events in the integrated plant logic model.

When identifying the CCCGs, the following guidelines were followed:

- Components within a CCF group have similar attributes and failure mechanisms, and are functionally redundant with respect to each other.

- Only the failure to perform a specific function is modelled in each CCF group.

- Low/consequential failure modes (e.g. spurious failures) are compared and excluded from common cause consideration if a more consequential active common cause group is modelled.

- The specific origins of common cause failure (e.g. shock, high temperature, manufacturing defects, etc.) are not specifically defined.

The five main coupling factor categories that were used in the identification of the CCCGs were [Ref-25.31]:

(1) Quality – Manufacturing quality,

(2) Design – Component design,

(3) Maintenance – Consideration of test and maintenance procedures, schedules and staff,

(4) Operation – Operational procedures, and

(5) Environment – External and internal environments.

This analysis was used to aid decision making when assigning CCCGs in addition to consideration of the component function.

A revision to the CCCGs may result if information not available now due to lack of design and manufacturing information becomes available.

The common cause groupings used in the UK ABWR PSA are generally limited within the same systems. This limitation is made for the following reasons:

- Common cause failures are considered for similar components (usually nearly identical) that operate in similar environments and have similar duty cycles. These characteristics are usually not present when considering components across system boundaries.

- It is generally found that the testing and maintenance procedures for different systems are such that a common cause failure of similar components among different systems is all but precluded.

The exception to this rule is the CCCG for the S/P strainers of all ECCS systems, due to high environmental similarity.

The alpha-factor method was used to quantify CCFs in the UK ABWR PSA. The alpha-factor method is preferred as it enables better modelling of higher Common Cause Component Groups (CCCGs) and more accurate modelling of uncertainty in the CCF parameters.

The alpha-factor method was not applied to the specific components such as digital C&I components, software errors and ECCS suction strainers in S/P.

25.4.8 Quantification Methodology

25.4.8.1 Introduction and Scope

The purpose of this section is to document the calculation of the Core Damage Frequency (CDF) due to events that occur when the plant is operating at Power. This section covers IEAP only. The calculation of CDF for external events and shutdown conditions is described in other sections.

As discussed earlier in this section, the UK ABWR PSA model consists of event trees and fault trees that are quantified using a fault tree linking process. The event trees are described in Section 25.4.3, while the fault trees are described in Section 25.4.4.

The calculation of the IEAP CDF is performed as a single top gate. The use of the sequence markers results in some non-minimal cutsets; however, this contribution is small and is judged not to significantly impact the insights.

Section 25.7 presents the results of the quantification.

(1) <u>Modelling Consistency</u>

Modelling consistency for the UK ABWR has been reviewed in each subtask, such as system fault tree and event tree modelling. The PSA team and the Engineering team had several meetings to exchange the latest information on design progress and development of the PSA model, and to confirm that the analysis correctly reflects the as-designed information. In addition, the dependency matrices developed during the identification and grouping of initiating events were reviewed in meetings between the PSA team and systems engineers.

Cutset review meetings were held between Hitachi-GE and GE-Hitachi. In these meetings, preliminary results with cutsets were reviewed (together with the Level 2 PSA results) to confirm that the PSA models are producing logically consistent results, that only minimal cutsets are being produced (no extraneous irrelevant failures in cutsets), and to the greatest extent possible confirm the accuracy of the model and its assumptions. The participants had expertise in PSA, system design, and plant operations.

Peer review and independent review were also conducted.

Through these meetings and follow up activities, modelling consistency and operational consistency were assured.

(2) Quantitative Definition of Significance in Internal Events at Power Level 1 PSA

This subsection describes the definition of significance in accident sequence, cutsets and basic events. The definitions are based on the ASME standard [Ref-25.13].

Significant accident sequence: one of the set of accident sequences resulting from the analysis of an Internal Events at Power (Level 1), defined at the functional or systematic level, that, when rank-ordered by decreasing frequency, sum to a specified percentage of the core damage frequency for Internal Events at Power, or that individually contribute more than a specified percentage of core damage frequency. The summed percentage is 95 percent and the individual percentage is 1 percent of IEAP CDF.

Significant cutset: one of the set of cutsets resulting from the analysis of an Internal Events at Power (Level 1), when rank ordered by decreasing frequency, sum to a specified percentage of the core damage frequency for CDF, or that individually contribute more than a specified percentage of core damage frequency. The summed percentage is 95 percent and the individual percentage is 1 percent of IEAP CDF.

Significant basic event: a basic event that contributes significantly to the computed risks for a specific hazard group. For internal events, this includes any basic event that has a Fussell-Vesely (F-V) importance greater than 0.005 or a Risk Achievement Worth (RAW) importance greater than 2.

25.4.8.2 Model Integration and Quantification Process

The purpose of the CDF quantification is to obtain the Boolean equation corresponding to the final event: "Core Damage". The quantification is developed in terms of Minimal Cutsets (MCS), which represent the minimal combinations of events that result in core damage.

The following key aspects characterise the CDF quantification process:

- Event trees to model plant response to each group of initiating events,
- Fault trees to model the behaviour of front-line and support systems,
- Integration of event tree and fault tree structures into a single linked model, and
- Quantification of the linked Boolean model with the probabilistic database and boundary condition files (flag files).

(1) PSA Software

The PSA development for IEAP Level 1 PSA was carried out using the CAFTA software, which has been developed by EPRI [Ref-25.32]. CAFTA is a part of the EPRI Risk and Reliability (R&R) Workstation suite of applications [Ref-25.33] and has been in use for over 22 years as a computer software program for developing reliability models of large complex systems, using both fault tree and event tree methodologies.

CAFTA has worldwide users at more than 70 power plants [Ref-25.34] and has a very large active user community ranging from power generation to communications, transportation, aviation and space applications. CAFTA was used during the previous GDA and also for the ABWR in US Design

Certification process. Based on this experience, CAFTA has been judged to be suitable for the UK ABWR PSA in GDA.

The EPRI R&R Workstation contains a large number of codes, which can be used in conjunction with CAFTA for PSA model development. This includes: FTREX for the PSA model quantification engine [Ref-25.35], PRAQUANT for event tree sequence by sequence quantification [Ref-25.36], FRANX for fire/flood analysis [Ref-25.37] and UNCERT for uncertainty analysis [Ref-25.38].

(2) <u>File Structure</u>

Event Tree Files

These files contain the event tree structures for each group of initiating events. The core damage sequences of these models are converted into fault tree logic and integrated into the Master Fault Tree File along with the system fault tree logic.

Fault Tree Files

These files contain the fault tree structures for each group of system fault trees.

Database File

The database contains the probabilities and frequencies of each of the events associated with the fault tree.

Quantification File

This file contains the conditions for quantifying the different accident sequences, or a single top gate that combines all the sequences. The quantification file identifies the following:

- Master fault tree file name,
- Database file name,
- Sequences to be quantified, or single top,
- Quantification truncation limits for each sequence, or single top,
- Flag files for each sequence, or a single master flag file (using selected configuration),
- Accident class for each sequence, and
- Mutually Exclusive file.

Flag Files

Flag files contain boundary conditions (for example: type of initiating event, assumed plant configuration) used in the quantification of the accident sequences. Binary model elements (that have either “True” or “False” values) called “flags” (also known as “house events”) are used to identify boundary conditions in the model structure. Flag files identify the flag events and associated binary values used in the quantification of the different accident sequences. A single master flag file is used to select the equipment line-up configuration.

"Mutually Exclusive" Files

A cutset file is used to identify combinations that should be excluded from the results. These mutually exclusive combinations identify and delete:

- Combinations of maintenance actions in multiple trains not allowed by technical specifications,
- Combinations of failures modes for equipment that are not possible (e.g. a specific basic event representing valve fails open combined with a basic event for the same valve failing closed), and
- Cutsets that are not appropriate (e.g. human failures that are only applicable for some sequences, but not others).

The mutually exclusive combinations are written as fault tree logic in a dedicated fault tree file.

This treatment excludes unacceptable combinations in operation by unplanned maintenance.

Recovery Files

Recovery files are not used in the IEAP Level 1 PSA model.

(3) <u>Steps for Model Integration</u>

The model integration included the following steps:

1. Define and group IEs (Section 25.4.1).
2. For the IE groups included in the quantification, identify all ETs developed for these IEs (Section 25.4.3).
3. Create a top logic fault tree from all ETs.
4. Merge the system model fault tree files into the master fault tree.
5. Review and modify (as needed) system model fault trees to include all system tops in the ETs.
6. Check for any model inconsistencies and fix the identified issues, including any issues with fault tree logic and reliability data.
7. Develop a mutually exclusive cutset file [Ref-25.14].

(4) <u>Steps for Quantification</u>

As discussed in the integration steps, the IEAP PSA quantification was developed with event tree / fault tree linking method.

The following steps were performed for model quantifications:

1. Identify and model credible recovery actions.
2. Review and (if needed) modify human dependency model.

3. Confirm the CDF has converged.

4. Perform cutset reviews with knowledgeable plant design/operation staff and PSA staff to identify any modelling issues.

5. Address the identified issues and repeat the model integration and quantification steps as needed.

6. Generate results for quantification and report the results in terms of sequences, IEs and etc. Report risk significances for basic events.

7. Identify significant contributors to CDF, such as initiating events, accident sequences, equipment failures, common cause failures, and operator errors. Include SSCs and operator actions that contribute to initiating events and event mitigation.

8. Perform uncertainty analysis using the code UNCERT within a cutset file (derived by single top quantification) and a reliability database with all uncertainty parameters, taking into account the state-of-knowledge correlation.

9. Review and identify sources of model uncertainties and assumptions used for quantification, and perform sensitivity studies.

(5) <u>Quantification Output</u>

Quantification of the model resulted in the following key outputs:

- Overall core damage frequency
- Core damage frequency as a function of
  - Initiating event
  - Accident sequence
  - Accident class
- Importance characterisation of individual events (in terms of industry standard risk importance measures, for example: Fussell-Vesely; Risk Achievement Worth, and so forth) relative to the core damage frequency

25.5 Level 2 PSA for Internal Events at Power

An objective of a Level 2 PSA is to provide the radioactive release frequencies (such as Containment Failure Frequency (CFF), Large Release Frequency (LRF), and Large Early Release Frequency (LERF) for reactor faults) and the associated source terms. Various modelling procedures are common in Level 1 PSA and Level 2 PSA. Such examples are data analysis and system analysis. Based on the main elements of a Level 2 PSA defined in IAEA TECDOC-1106 [Ref-25.57], Level 2 specific tasks are:

- Interface between Level 1 and Level 2 (Section 25.5.1),
- Severe Accident Analysis (Section 25.5.2.2),
- Containment Performance Analysis (Section 25.5.2.3),
- Quantification Method (Section 25.5.2.4), and
- Source Term Analysis (Section 25.5.3).

Note: The parentheses above are the section number in this report.

Figure 25.5-1 shows the overview of Level 2 PSA process.

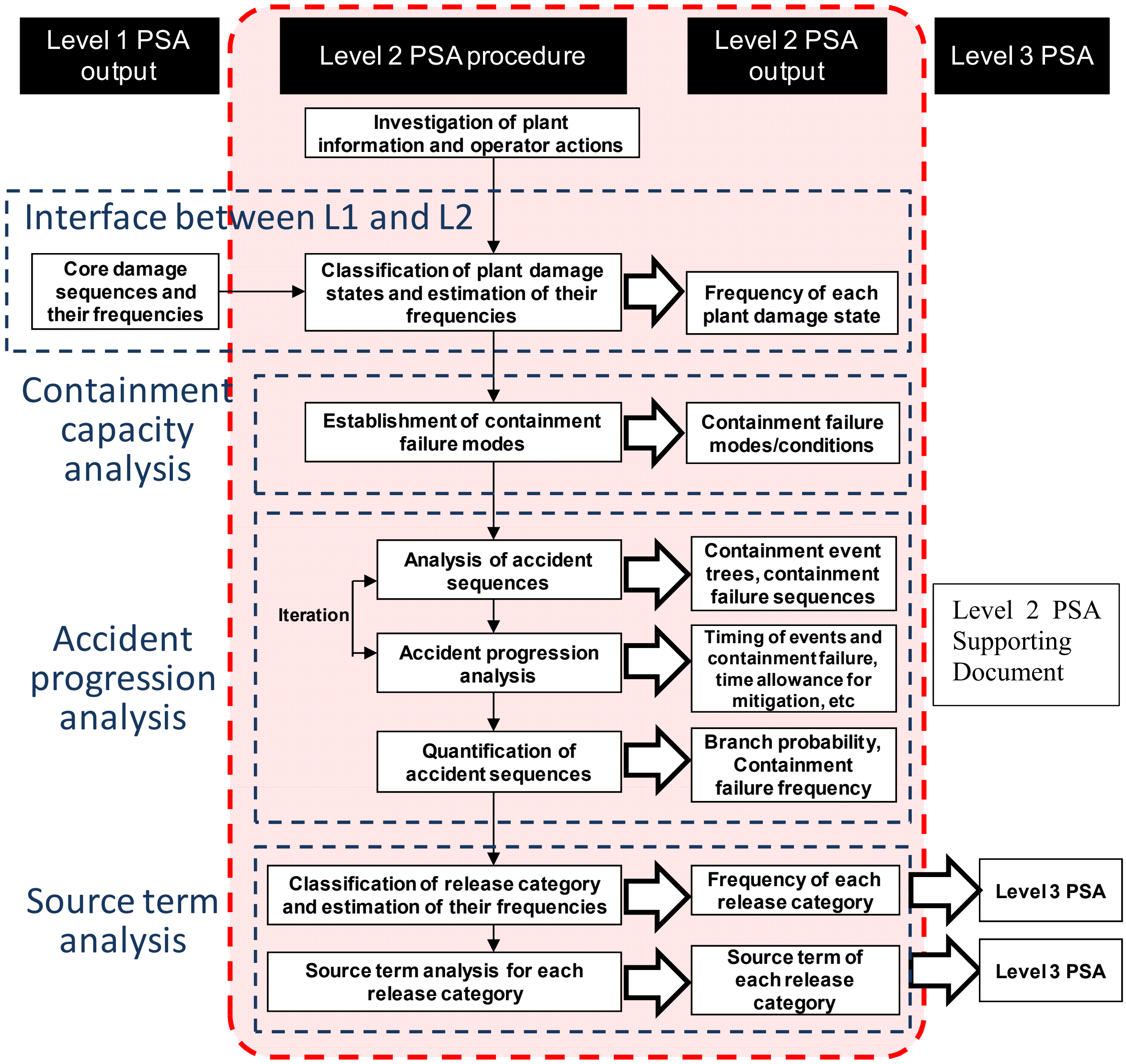

Figure 25.5-1 Overview of Level 2 PSA Process

25.5.1 Interface between Level 1 and Level 2

In the Level 1 PSA, accident classes are defined as the end state. To address the characteristics of a Level 2 accident progression such as the reactor pressure and timing of core damage, system status after core damage and status of containment and containment safeguards, this section discusses the interface between Level 1 and Level 2, which is defined as PDSs (Plant Damage States). PDS grouping focuses on the key parameters that would influence the containment responses and source terms. The detail of PDS grouping is described in the Topic Report on internal events at power Level 2 PSA [Ref-25.10].

25.5.1.1 Approach on PDS

Accident classes have been categorised into twenty-six groups. These accident classes are taken over as PDSs in the Level 2 PSA. Figure 25.5.1-1 shows the logic tree of PDS grouping. The accident sequences are grouped into several PDSs based on some of the key items described below. These key items are selected in order to use the PDSs as the interface between Level 1 and Level 2 PSA. Therefore, the PDSs are used as initial conditions of the Containment Event Trees (CETs).

(1) RPV pressure at core damage

Reactor Pressure Vessel (RPV) pressure at the time of vessel failure following core damage affects the potential for Direct Containment Heating (DCH) to cause containment failure coincident with vessel failure.

RPV depressurisation after core damage is credited considering the time margin to RPV failure even if RPV depressurisation has failed before core damage. RPV pressure also has a relationship with the availability of low pressure injection systems.

(2) Timing of core damage

The timing of core damage influences the feasibility of manual actions to mitigate severe accidents and the Primary Containment Vessel (PCV) conditions such as pressure and temperature. In addition, the timing of core damage has a relationship with the timing of fission product release which would influence the consequence analysis (Level 3 PSA).

(3) Availability of molten debris cooling / heat removal measures

If debris cooling measures such as ECCS and FLSS are available, molten debris cooling can be achieved. This could potentially maintain RPV to be intact, which in turn could prevent the occurrence of DCH, ex-vessel Fuel Coolant Interaction (FCI) or Molten Core Concrete Interaction (MCCI). If heat removal measures such as RHR or PCV vent are available, the containment overpressure can be mitigated. Availabilities of these measures depend on availabilities of support systems such as AC and/or DC power.

(4) Timing of containment failure

If PCV fails, equipment in reactor building would fail due to the adverse impacts of steam. Therefore, the timing of containment failure has a relationship with the availability of some mitigation measures.

(5) Availability of Containment Isolation

The failure of containment isolation at the time of core damage would result in an early fission product release. In case of ISLOCA or BOC event, isolation of failed piping has already been examined in the Level 1 PSA. The dependency of this failure is considered in the Level 2 PSA.

(6) Suppression Pool Bypass

The last key item for PDS grouping is the status of suppression pool. This item influences the steam condensation effect by the suppression pool and the scrubbing effect of fission products.

25.5.1.2 PDS Definition

As shown in Table 25.5.1-1 and Figure 25.5.1-1, twenty-six PDSs were identified based on the key items introduced in the previous section. Identifiers for defining the name of each PDS are listed in Table 25.4.3-1. The PDSs are briefly described below. As a result of PDS grouping, PDSs in Level 2 are the same as the accident classes in Level 1.

(1) S4

This PDS is an excessive LOCA (RPV rupture) inside containment. Core damage occurs directly in the early phase with the RPV at low pressure. After core damage has occurred, availability of debris cooling and heat removal measures are treated in a probabilistic way in the CET. Failure of the PCV is assumed to occur with the initiating event.

(2) AE

This PDS includes AE, S1E and S2E which result in RPV low pressure at core damage. E means a failure of water injection measures, and A, S1, and S2 denotes the break size of LOCA. This PDS means LOCA with a failure of water injection measures. In this sequence, core damage is occurred in early phase because water inventory in the RPV flows out to the containment. After core damage, availabilities of debris cooling measures and heat removal measures are treated in probabilistic way in the CET.

(3) TQUV

The PDS “TQUV” is a Transient (T) including planned manual shutdown and special initiators, followed by failures of feedwater system (Q), high pressure ECCS (U) and low pressure ECCS including alternative water injection system (V). The success of RPV depressurisation results in low pressure in short term. Failure of depressurisation of the RPV is treated in other PDSs such as TQUX. The PDS is categorised in the low pressure and early core damage case. In the Level 2 PSA, the availabilities of systems for the debris cooling and heat removal measures are treated in probabilistic approach under the failure of low pressure ECCS. For example, the PCV spray system shares the pumps and valves with the low pressure ECCS. The dependency is treated in the CET.

(4) TQUX

The PDS “TQUX” is a Transient (T) including planned manual shutdown and special initiators, followed by failures of feedwater system (Q), high pressure ECCS (U) and reactor depressurisation (X). Since high pressure injection measures are lost and reactor depressurisation fails, core damage occurs at high pressure in short term. In this case, the low pressure ECCS is available.

(5) S12UX

The PDS “S12UX” is a LOCA initiator in the RPV drain line, followed by failures of high pressure injection (U) and reactor depressurisation (X). Since high pressure injection measures are lost and reactor depressurisation fails, core damage occurs at high pressure in short term. The LOCA does not depressurise the RPV prior to breach of the RPV. In this case, because the low pressure ECCS can be assumed to be available, the low pressure ECCS for debris cooling and heat removal after the RPV failure due to molten debris is credited.

(7) TB

The PDS “TB” is a Loss of Offsite Power Transient (T) including planned manual shutdown and special initiators, followed by a loss of AC power for engineered safety systems (B). Therefore, TB means “Station Blackout”. For the UK ABWR, TB sequences are further divided into four PDSs (TB, TBU, TBD and TBP) from the viewpoint of PCV response. Availability of the debris cooling and heat removal measures is treated in a probabilistic way, since it depends on recovery probabilities of AC / DC powers.

(8) TB

TB is one of the TB sequence groups. It also includes failures to recover offsite power by 30 minutes and 8 hours. Since the UK ABWR has RCIC as high pressure ECCS and RCIC can inject water to the core without AC power, the core can be covered by water until DC power for the RCIC flow rate control is depleted. The timing of core damage is long term because of the water injection of the RCIC. Since the depletion of DC power also disables manual depressurisation by opening SRVs as well as RCIC operation, the RPV pressure is high at the moment of core damage.

(9) TBU

TBU is one of the TB sequences groups, which is followed by a failure of high pressure ECCS (U). TBU also includes failure to recover offsite power by 30 minutes. All the water injection systems which need AC power are unavailable, and RCIC, which can inject water to the core without AC power, fails due to various causes such as mechanical failure. In this case, depressurisation is possible because DC power is available. However, the RPV depressurisation action is not conducted because there are no low pressure water injection measures for low pressure conditions even if the RPV is successfully depressurised. TBU core damage occurs in short term at RPV high pressure.

(10) TBD

TBD is one of the TB sequence groups, which is followed by a loss of DC power for engineered safety systems (D). Both AC and DC powers are unavailable in this sequence. Therefore, all of water injection systems and all of heat removal systems are not available. It results in the core damage that occurs in short term at high RPV pressure.

(11) TBP

TBP is one of the TB sequence groups, which is followed by failure to reclose SRV (P). Reactor depressurisation via SRV opening impedes injection of water to the core by RCIC because RCIC uses high pressure steam from RPV as its driving force. Since ECCS is also disabled due to loss of AC power, core damage occurs in short term at RPV low pressure.

(12) TW

The PDS “TW” is a Transient (T) including planned manual shutdown and special initiators, followed by failure of heat removal measures from the PCV (W). In TW sequences, high pressure injection to the core is successful and PCV heat removal fails. The core can be covered by water for a long period. However, decay heat generated from the core cannot be removed from the PCV, and the PCV is finally failed by overpressure due to decay heat accumulation. Failures of equipment installed in Reactor Building are assumed due to the adverse effect of steam at the time of the PCV failure. This treatment is conservative because the failure location due to overpressure is considered as the Drywell (D/W) top head, and steam might not affect the safety systems such as RHR, electric devices and so on. In the Level 2 PSA, RHR (LPFL) located in R/B is credited. However, RHR has failed in TW sequences. Therefore this conservatism should have a small impact on LRF. As a result, the PCV fails due to overpressure first and then, it results in core damage at RPV high pressure due to loss of water injection to the core. The Level 1 PSA is performed from the plant initiating events to core damage but the PCV has already failed. Therefore, containment event tree analysis after Level 1 PSA is conducted and severe accident analysis by MAAP code [Ref-25.46] is conducted for a representative scenario because the TW sequences have to be considered as fission product releases.

(13) S12W

The PDS "S12W" is a LOCA initiator in the RPV drain line, followed by failure of heat removal measures from the PCV (W). It means that water injection to the core is successful and heat removal from the PCV fails. The core can be covered by water for a long period. However, decay heat generated from the core cannot be removed from the PCV, and the PCV fails by overpressure due to decay heat accumulation. Failures of all water injection measures to the core are assumed to occur at the time of the PCV failure because the equipment for the water injection is impacted by the PCV failure. As a result, the PCV fails due to overpressure first and then, it results in core damage at high pressure due to the loss of water injection to the core.

(14) TW-LP

The PDS "TW-LP" is a Transient (T) including planned manual shutdown and special initiators, followed by failure of heat removal measures from the PCV (W). It means that the RPV is depressurised and low pressure injection to the core is successful and heat removal from the PCV fails. The core can be covered by water for a long period. However, decay heat generated from the core cannot be removed from the PCV, and the PCV fails by overpressure due to decay heat accumulation. All water injection measures to the core are assumed to fail at the time of the PCV failure. As a result, PCV fails due to overpressure first and then, it results in core damage at low pressure due to the loss of water injection to the core.

(15) AW-LP

The PDS "AW-LP" is a LOCA sequence, followed by failure of heat removal measures from the PCV (W). It means that the RPV is depressurised and low pressure injection to the core is successful and heat removal from the PCV fails. The core can be covered by water for a long period. However, decay heat generated from the core cannot be removed from the PCV, and the PCV fails by overpressure due to decay heat accumulation. Failures of all water injection measures to the core are assumed at the time of the PCV failure. This treatment is conservative because the most probable failure location due to overpressure is at the drywell top head, and steam might not affect the safety systems such as RHR, electric devices and so on. In the Level 2 PSA, RHR (LPFL) located in R/B is credited. However, RHR has failed in AW-LP sequences. Therefore this conservatism should have a small impact on LRF. As a result, PCV fails due to overpressure first and then, it results in core damage at low pressure due to the loss of water injection to the core.

(16) AC

The PDS "AC" is a LOCA sequence with low RPV pressure (including consequential LOCA) with failure of reactivity control. PCV failure occurs early and it results in core damage at low pressure.

(17) TC

The PDS "TC" is a Transient (T) including planned manual shutdown and special initiators, followed by the failure of reactivity control (C). In this PDS, water injection to the core is successful but both control rod insertion and boric acid injection into the core are failed. Heat removal from the PCV is not sufficient to keep the core cooled because core power of this case is kept relatively higher than that of decay heat. As a result, PCV overpressure failure occurs earlier than the TW sequence. After that, it results in core damage at high pressure due to loss of water injection to the core by the same reason as TW sequence.

(18) S12C

The PDS "S12C" is a liquid LOCA initiator, with failure of reactivity control. PCV failure occurs early and it results in core damage at high pressure.

(19) TC-HP

The PDS "TC-HP" is a Transient (T) including planned manual shutdown and special initiators, with failure of control rod insertion but boric acid injection into the core is successful. High pressure injection fails and the RPV is not depressurised resulting in core damage at high pressure due to loss of water injection to the core.

(20) TC-LP

The PDS "TC-LP" is a Transient (T) including planned manual shutdown and special initiators, with failure of control rod insertion but boric acid injection into the core is successful. In this PDS, high pressure injection to the core fails and the RPV is depressurised but low pressure injection fails. As a result, core damage occurs at low pressure due to loss of water injection to the core.

(21) S3E

This PDS is a LOCA outside containment. E means a failure of water injection measures, and S3 means a LOCA outside containment. Core damage occurs directly in the early phase with the RPV at low pressure. After core damage has occurred, availability of debris cooling and heat removal measures are treated in a probabilistic way in the CET.

(22) S3UX

This PDS is a LOCA outside containment, followed by failures of high pressure injection (U) and reactor depressurisation (X). Core damage occurs directly in the early phase with the RPV at high pressure. After core damage has occurred, availability of debris cooling and heat removal measures are treated in a probabilistic way in the CET.

(23) TNQUV

The PDS "TNQUV" is a Transient (T) with an SRV tailpipe break in Wetwell (W/W) airspace, followed by failures of feedwater system (Q), high pressure ECCS (U) and low pressure ECCS (V). The success of RPV depressurisation results in low pressure in short term. This PDS is categorised as low pressure core damage with S/P bypass. The PCV fails before core damage due to overpressure when RPV is depressurised with SRVs.

(24) TBPN

TBPN is one of the TB sequence groups with the loss of Class 1 and Class 2 AC power and an SRV tailpipe break in W/W airspace. Since ECCS is also disabled due to loss of AC power, core damage occurs in the early phase at high pressure with S/P bypass.

(25) TNQUX

The PDS "TNQUX" is a Transient (T) with an SRV tailpipe break in W/W airspace, followed by failures of feedwater system (Q), high pressure ECCS (U) and reactor depressurisation (X). Since high pressure injection measures are lost and reactor depressurisation fails, core damage occurs at high pressure in short term with S/P bypass.

(26) TBDN

TBDN is one of the TB sequence groups with the loss of Class 1 and Class 2 AC/DC power and an SRV tailpipe break in W/W airspace. Therefore, all water injection systems and all heat removal systems for engineered safety systems are not available. Since all water injection systems are unavailable due to loss of AC / DC powers, core damage occurs in the early phase at high pressure with S/P bypass.

(27) TCN

The PDS "TCN" is a Transient (T), followed by a failure of control rod insertion and an SRV tailpipe break in W/W airspace. This PDS results in containment failure in short term followed by core damage at high pressure with S/P bypass.

(28) AN

This PDS "AN" is a LOCA sequence (all LOCAs including consequential LOCA) with the failure of W/W to D/W vacuum breakers to open and the failure of RPV injection. This results in early PCV failure and core damage at low pressure.

The above PDSs are divided further based on availability of AC/DC power in order to develop the sequence specific fault trees in CETs. Tables 25.5.1-2 and 25.5.1-3 show the list of CETs developed for this Level 2 PSA. This treatment is described in Section 3.2 of the Topic Report on internal events at power Level 2 PSA [Ref-25.10].

25.5.1.3 Treatment of specific issues

(1) Dependency Treatment

Dependencies between Level 1 PSA and Level 2 PSA can be grouped into three categories. Each category has been addressed in the Level 2 PSA as shown below.

- Scenario dependencies
- Equipment failures prior to core damage including common cause failure
- Operator action dependencies

Scenario dependencies

Headings in Level 2 PSA may depend on scenario specific conditions such as RPV pressure, vapour suppression system (VSS) failure, and LOCA in Level 1 PSA. For example, the heading of "PCV failure due to DCH" does not need to be considered in case of the PDSs with low RPV pressure at vessel failure. All scenario dependencies between Level 1 and Level 2 are considered in the CET for each PDS.

Equipment failures prior to core damage including common cause failure

The CAFTA computer code [Ref-25.32] was used to create fault trees for the CETs. The code placed the gate for each PDS under the OR gate for the failure paths to ensure all equipment failure linkages were included. Each fault tree top from the Level 1 PSA was merged to the gate of the PDS. By linking the fault trees of Level 1 PSA to those of Level 2 PSA, dependencies regarding basic events including component, support system, operator action and success logic ("NOT" logic) were captured in the analysis. In addition, the system dependencies for each CETs were considered.

Operator action dependencies

Three types of emergency operation strategies, measures and procedures to prevent or mitigate fission product release were considered:

- Core cooling strategy,
- Containment control strategy, and
- Fission product release control strategy.

The core cooling strategy is to depressurise the RPV and inject water to the damaged core using any available injection systems. This operation is similar to core cooling credited in Level 1 PSA. Therefore operator action dependencies between Level 1 PSA and Level 2 PSA were considered.

Note: Offsite power recovery is considered when all AC power is lost in Level 2 PSA.

The containment control strategy is to cool debris and reduce PCV temperature/pressure by PCV spray or PCV venting using any available injection systems. The dependency of operator action for containment control strategy is considered in Level 2 PSA.

The fission product release control strategy is to conduct PCV venting and manage hydrogen concentration in the Reactor Building (R/B). Hydrogen management in R/B is not explicitly considered in this Level 2 PSA. If hydrogen combustion occurs in the reactor building, the roof of the reactor building might fail due to the dynamic load of hydrogen combustion as occurred in the Fukushima accident.

The analysis condition for the source term where blowout panel successfully open gives more severe result than the analysis condition where blowout panel does not open. In addition, even if PCV failure occurs and blowout panel does not open, hydrogen concentration in the R/B does not reach the flammable limit in a short time because containment is inerted with nitrogen and a large amount of steam is generated during the severe accident at power. Therefore, the blowout panel is conservatively assumed to open successfully at the design pressure in terms of the source term, and the condition of the opening installed in the refueling floor is not considered in the IEAP Level 2 PSA because the release fraction of fission products is dominated by the opened R/B blowout panel.

(2) Containment De-inert operation

The containment is inerted about 16 hours after the start of control rod withdrawal (start of period covered by at power PSA) for the purpose of drywell inspection just after the reactor has achieved rated pressure. In addition, the containment is de-inert about 20 hours before the breaking of condenser vacuum (end of period covered by at power PSA) for the purpose of drywell inspection when the reactor is still at nominal rated pressure.

The total time (16 hours plus 20 hours) at which the containment is not inerted is less than 0.3 percent of the duration of an operation cycle (18 months). The conditional containment failure probability (CCFP) is higher during de-inert period than inerted period due to the potential for hydrogen combustion.

Meanwhile, there may be planned or forced outages during a cycle. In this case, de-inert duration would increase. This sensitivity analysis is in Section 25.7.

To address the risk due to de-inert operation, it is assumed that all core damage events during de-inert operation lead to containment failure and loss of all mitigation features. This risk is treated as a "bypass" containment failure mode.

The risk of SFP structural failures due to hydrogen combustion in R/B (reactor challenge) is evaluated in SFP PSA.

Table 25.5.1-1 PDS Characteristics (1/2)

<table>
<tr><th>PDS</th><th>RPV pressure at core damage[8]</th><th>Timing of core damage[9]</th><th>Debris cooling / heat removal measures</th><th>Containment integrity</th><th>Containment Isolation</th><th>S/P bypass</th></tr>
<tr><td>S4</td><td rowspan="3">Low</td><td rowspan="5">Short</td><td>LPFL/RHR[1], FLSS[1], FLSR[1], LDF</td><td rowspan="9">Examined in Level 2 PSA</td><td rowspan="9">Examined in Level 2 PSA</td><td rowspan="12">No</td></tr>
<tr><td>AE</td><td>LPFL/RHR[2], FLSS[1,2], FLSR[1], LDF</td></tr>
<tr><td>TQUV</td><td>LPFL/RHR[2,3], FLSS[2,3], FLSR[1],LDF</td></tr>
<tr><td>TQUX</td><td rowspan="4">High</td><td rowspan="2">LPFL/RHR[1], FLSS[1], FLSR[10], LDF</td></tr>
<tr><td>S12UX</td></tr>
<tr><td>TB</td><td>Long</td><td rowspan="3">LPFL/RHR[1], FLSR[10], LDF</td></tr>
<tr><td>TBU</td><td rowspan="3">Short</td></tr>
<tr><td>TBP</td><td>Low</td></tr>
<tr><td>TBD</td><td rowspan="3">High</td><td>LDF , FLSR[10]</td></tr>
<tr><td>TW</td><td rowspan="4">Long</td><td rowspan="7">FLSS[1,2], FLSR[10], LDF[7]</td><td rowspan="4">Failed prior to core damage (overpressure, long term)</td><td rowspan="7">Not defined (containment already failed)</td></tr>
<tr><td>S12W</td></tr>
<tr><td>TW-LP</td><td rowspan="3">Low</td></tr>
<tr><td>AW-LP</td><td rowspan="2">Yes[4]</td></tr>
<tr><td>AC</td><td rowspan="2">Short</td><td rowspan="3">Failed prior to core damage (overpressure, short term)</td></tr>
<tr><td>TC</td><td rowspan="3">High</td><td rowspan="4">No</td></tr>
<tr><td>S12C</td><td>Long</td></tr>
<tr><td>TC-HP</td><td rowspan="2">Short</td><td>LPFL/RHR[2], FLSS[1], FLSR[10], LDF</td><td rowspan="2">Examined in Level 2 PSA</td><td rowspan="2">Examined in Level 2 PSA</td></tr>
<tr><td>TC-LP</td><td>Low</td><td>LPFL/RHR[2,3], FLSS[1], FLSR[10], LDF</td></tr>
</table>

Table 25.5.1-1 PDS Characteristics (2/2)

<table>
<tr><th>PDS</th><th>RPV pressure at core damage[8]</th><th>Timing of core damage[9]</th><th>Debris cooling / heat removal measures</th><th>Containment integrity</th><th>Containment Isolation</th><th>S/P bypass</th></tr>
<tr><td>S3E</td><td>Low</td><td rowspan="8">Short</td><td rowspan="2">FLSS[1,2], LDF</td><td rowspan="2">Bypass[6]</td><td rowspan="2">Examined in Level 2 PSA[6]</td><td rowspan="2">Yes</td></tr>
<tr><td>S3UX</td><td>High</td></tr>
<tr><td>TNQUV</td><td>Low</td><td>FLSS[2,3], LDF</td><td>Failed prior to core damage (overpressure, short term)</td><td>Not defined (containment already failed)</td><td rowspan="5">Yes[5]</td></tr>
<tr><td>TBPN</td><td rowspan="4">High</td><td rowspan="2">LPFL/RHR[1], FLSS[1], LDF</td><td rowspan="3">Examined in Level 2 PSA</td><td rowspan="3">Examined in Level 2 PSA</td></tr>
<tr><td>TNQUX</td></tr>
<tr><td>TBDN</td><td>LDF</td></tr>
<tr><td>TCN</td><td>FLSS[1], LDF[7]</td><td>Failed prior to core damage (overpressure, short term)</td><td rowspan="2">Not defined (containment already failed)</td></tr>
<tr><td>AN</td><td>Low</td><td>FLSS[1,2], LDF[7]</td><td>Failed prior to core damage (sub-atmospheric overpressure, short term)</td><td>Yes[4]</td></tr>
</table>

Notes to Table 25.5.1-1

[1]Not examined in Level 1 PSA

[2]May have failed in Level 1 PSA, but examined probabilistically in Level 2 PSA (e.g. RPV injection valve fail to open in Level 1 PSA but spray line still available in Level 2 PSA, SBO in Level 1 PSA but offsite power recovered in Level 2 PSA)

[3]May have failed in Level 1 PSA due to human error(s), but examined probabilistically in Level 2 PSA using conditional human error probability following human error(s) in Level 1 PSA

[4]Fission products are released from RPV through drywell to PCV failure location (assumed to be in drywell).

[5]Fission products are released from RPV through broken SRV tailpipe to wetwell airspace.

[6]Manual isolation of BOC is not credited.

[7]It is assumed S/P drainage by LDF does not occur given containment failure in Level 1 PSA sequences.

[8]RPV pressure at the time of vessel failure affects the potential for Direct Containment Heating (DCH) to cause containment failure coincident with vessel failure. The success criterion of RPV depressurisation for DCH is lower than 2.0 MPa [gauge] [Ref-25.47].

[9]Timing of core damage reflects the immediate loss of core cooling versus later failure of injection, and affects possibility of containment pressurisation prior to the onset of core damage. PDS is provided with the results of the accident sequence analysis. Nonetheless, the criterion of timing of core damage is assumed to 4 hours with the reference of the definition of early release of PSA Applications Guide by EPRI [Ref-25.87].

[10]Time window analysis for FLSR is performed [Ref-25.10].

Table 25.5.1-2 CETs Considering Availability of AC/DC Power (1/2)

No.	PDS	CET	Availability of AC/DC power
1	AC	AC	AC power is available
2	AC	AC_LOOPA	Loss of offsite power
3	AE	AE	AC power is available
4	AE	AE_LOOPA	Loss of offsite power
5	AE	AE_C1SBO	Class 1 AC power is not available
6	AN	AN	AC power is available
7	AW-LP	AW-LP	AC power is available
8	AW-LP	AW-LP_LOOP	Loss of offsite power
9	S12C	S12C	AC power is available
10	S12UX	S12UX	AC power is available
11	S12UX	S12UX_LOOPA	Loss of offsite power
12	S12W	S12W	AC power is available
13	S12W	S12W_LOOPA	Loss of offsite power
14	S3E	S3E	AC power is available
15	S3E	S3E_LOOPA	Loss of offsite power
16	S3UX	S3UX	AC power is available
17	S3UX	S3UX_LOOPA	Loss of offsite power
18	S4	S4	AC power is available
19	TB	TB	Offsite power is recovered prior to core damage
20	TB	TB_C1SBO	Class 1 and Class 2 AC power is not available
21	TBD	TBD	DC power is not available
22	TBDN	TBDN	DC power is not available (VSS failure)
23	TBP	TBP	Offsite power is recovered prior to core damage
24	TBP	TBP_C1SBO	Class 1 and Class 2 AC power is not available

Table 25.5.1-2 CETs Considering Availability of AC/DC Power (2/2)

No.	PDS	CET	Availability of AC/DC power
25	TBPN	TBPN	Class 1 and 2 AC power is not available
26	TBU	TBU	Offsite power is recovered prior to core damage
27	TBU	TBU_C1SBO	Class 1 and Class 2 AC power is not available
28	TC	TC	AC power is available
29	TC	TC_LOOPA	Loss of offsite power
30	TC-HP	TC-HP	AC power is available
31	TC-HP	TC-HP_LOOPA	Loss of offsite power
32	TC-LP	TC-LP	AC power is available
33	TC-LP	TC-LP_LOOPA	Loss of offsite power
34	TCN	TCN	AC power is available
35	TCN	TCN_LOOPA	Loss of offsite power
36	TNQUV	TNQUV	AC power is available
37	TNQUX	TNQUX	AC power is available
38	TNQUX	TNQUX_LOOPA	Loss of offsite power
39	TNQUX	TNQUX_C1SBO	Class 1 AC power is not available
40	TQUV	TQUV	AC power is available
41	TQUV	TQUV_LOOPA	Loss of offsite power
42	TQUV	TQUV_C1SBO	Class 1 AC power is not available
43	TQUX	TQUX	AC power is available
44	TQUX	TQUX_LOOPA	Loss of offsite power
45	TQUX	TQUX_C1SBO	Class 1 AC power is not available
46	TW	TW	AC power is available
47	TW	TW_LOOPA	Loss of offsite power
48	TW-LP	TW-LP	AC power is available
49	TW-LP	TW-LP_LOOPA	Loss of offsite power

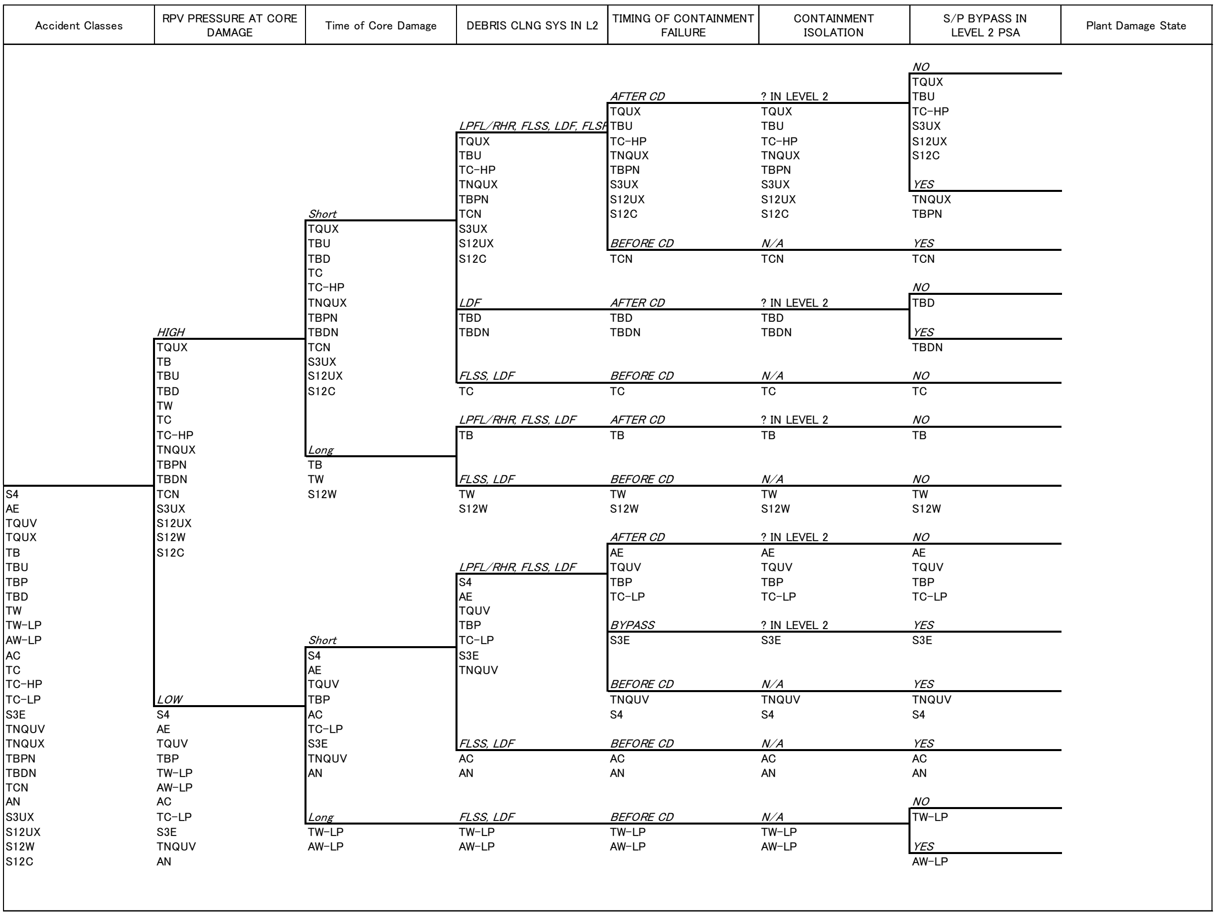

Figure 25.5.1-1 Grouping of Plant Damage States (PDSs)

25.5.2 Containment Event Tree and Quantification

This section provides the development of Containment Event Trees (CETs) and quantification method. The steps to quantify CETs are shown below.

- Development of CET structures such as headings and end states
- Evaluation of branch probability for phenomenological events
- Development of fault trees

The CETs are developed considering the success criteria and severe accident progressions in line with the results of the severe accident analyses and containment performance analysis. The detail of CETs development and quantification is described in the Topic Report on internal events at power Level 2 PSA [Ref-25.10].

25.5.2.1 Approach to Probability Derivation

(1) Containment Event Tree

Identification of Containment Failure Mode

Potential mechanisms that could challenge containment integrity in severe accident are identified to conduct the Level 2 PSA assessment. From the containment failure modes and general Accident Management (AM) guide, a summary of severe accident progression of ABWR is outlined in Figure 25.5.2-1. The accident progression scenario is divided into three phases: T1, T2 and T3. T1 is the phase before the RPV failure. T2 is the phase immediately after the RPV failure. T3 is the late phase of the severe accident.

A box with a red outline shows each PCV failure mode due to different severe accident phenomena in Figure 25.5.2-1. PCV failure due to overpressure occurs before core damage in sequences such as TW and the TCs. In-vessel FCI might occur when the corium falls into the pool of the RPV lower plenum (so called α-mode failure). Failure of the RPV might give a potential of ex-vessel FCI in the lower D/W, DCH with RPV failure in high pressure, and shell attack, which means direct heating of the containment by the debris. PCV failure by MCCI might occur if debris cooling is not sufficient. Subsequently, it might lead to loss of integrity of base structure of the containment. PCV pressure and PCV temperature are necessary to be controlled within ultimate capacity even after avoiding the above potential failure mechanisms.

Other mechanisms mentioned in IAEA SSG-4 [Ref-25.48] as examples of areas of uncertainty relevant to the progression of severe accidents and containment failures are:

- In-vessel hydrogen generation,
- In-vessel natural circulation,
- In-vessel fuel–coolant interactions (energetic and non-energetic),
- Failure mechanisms of reactor pressure vessel,
- High pressure melt ejection and/or Direct Containment Heating,
- Ex-vessel fuel–coolant interactions (energetic and non-energetic),
- Core–concrete interactions, and
- Hydrogen combustion.

Table 25.5.2-1 shows the consideration of the above areas for the UK ABWR Level 2 PSA.

Through the examination of accident progression in Figure 25.5.2-1 and comparison with existing studies, such as IAEA SSG-4 [Ref-25.48], the containment failure modes were identified and reviewed in Section 5 of the Topic Report on internal events at power Level 2 PSA [Ref-25.10].

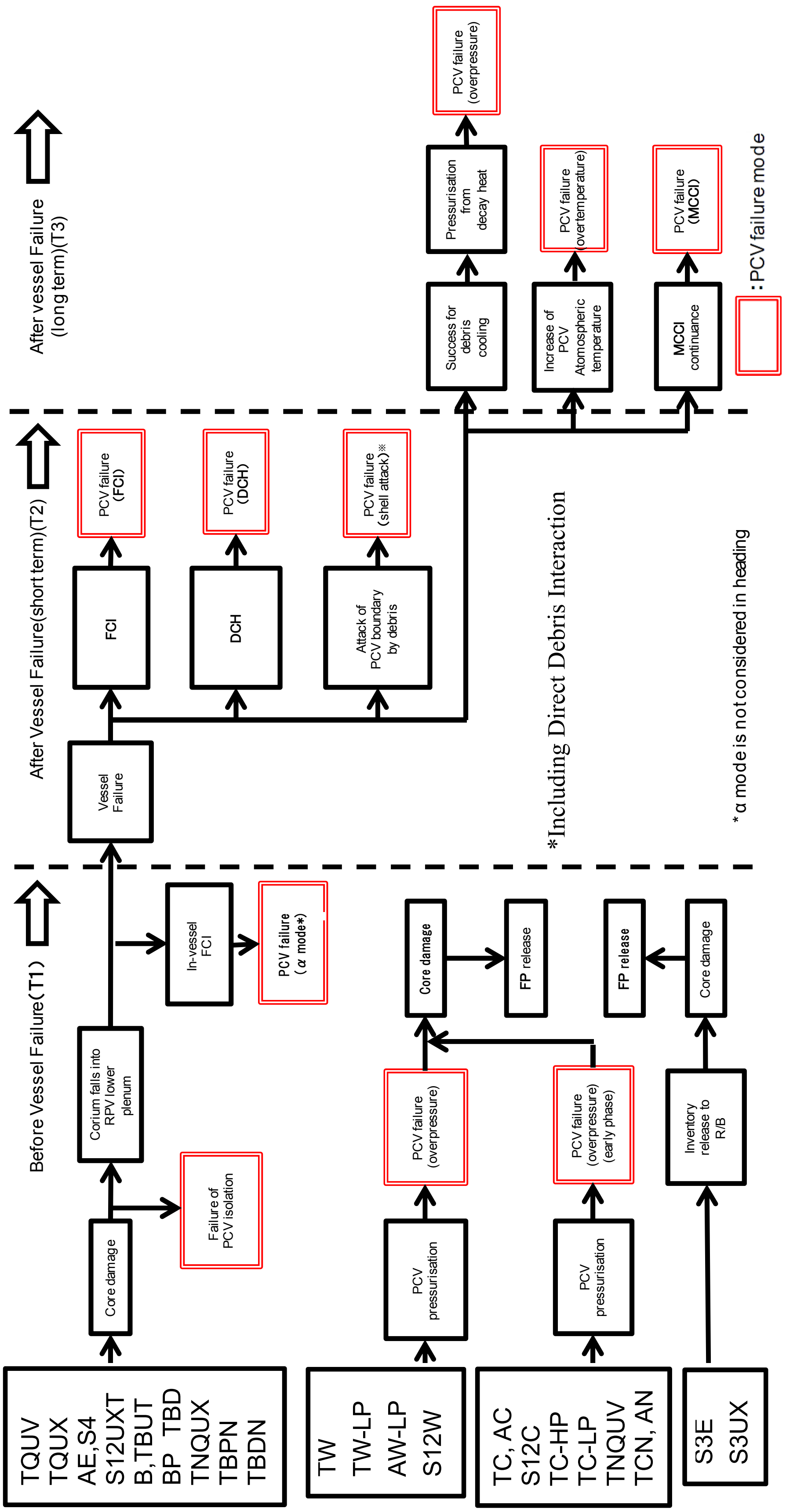

Figure 25.5.2-1 Overview of Severe Accident Progression and PCV Failure Mode for UK ABWR

Table 25.5.2-1 Type of Severe Accident Event and Consideration in Level 2 PSA Model

Type of severe accident event [Ref-25.48]	Consideration in Level 2 PSA model [Ref-25.10]
In-vessel hydrogen generation	Hydrogen from metal-water reaction is exhausted through the SRVs and pressurises the PCV.
In-vessel natural circulation	This issue is principally a PWR concern. Heat up and creep rupture in primary loop (e.g. hot leg nozzle, pressuriser surge line and steam generator tube) causing depressurisation in a PWR can have sub-atmospheric consequences on the accident progression. In contrast, depressurisation of RPV in a BWR has positive effect from the view point of severe accident control by allowing use of low pressure injection systems and precluding HPME.
In-vessel fuel–coolant interactions (energetic and non-energetic)	In-vessel FCI and steam explosion leading to RPV failure is not likely due to saturated condition of water and large volume fraction of structures in lower plenum. Nonetheless, In-vessel FCI is explicitly modelled in the CET to represent the phenomena and accident sequence.
Failure mechanisms of reactor pressure vessel	Failure mechanisms of RPV are considered in severe accident analysis code (MAAP) which evaluates the time margin for RPV failure.
High pressure melt ejection (HPME) and/or Direct Containment Heating (DCH)	Containment failure caused by DCH following HPME is considered in early failure CET heading and a branch probability is estimated using information in NUREG/CR-4700 [Ref-25.50]. The value of branch probability is supported by using a continuous spectrum formed probability distribution.
Ex-vessel fuel–coolant interactions (energetic and non-energetic)	Ex-vessel FCI causing containment failure is considered in CET heading and branch probability is estimated using information in NUREG/CR-4700 [Ref-25.50]. The value of branch probability is reviewed through the calculation by using of continuous spectrum formed probability distribution.
Core–concrete interactions	Containment failure caused by MCCI is considered in CET heading and branch probability is estimated using information in NUREG/CR-4700 [Ref-25.50]. The value of branch probability is reviewed through the calculation by using of continuous spectrum formed probability distribution.
Hydrogen combustion	The Atmospheric Control system establishes inert atmosphere in PCV at power condition. The inert atmosphere eliminates the potential for hydrogen combustion in the containment. De-inert operation is permitted for brief periods before and after outages. It is conservatively assumed that all core damage events during this de-inert operation lead to containment failure. This risk is treated as the “bypass” containment failure mode.

Development of Containment Event Tree

Based on information from Sections 25.5.1 and “Identification of Containment Failure Mode” above, CETs have been developed for each PDS. Each CET is shown in Appendix B of the Topic Report on internal events at power Level 2 PSA [Ref-25.10].

As an example, a summary of TQUX CET and its development for the at Power Level 2 PSA is provided as shown in Figure 25.5.2-2.

TQUX is the one of the PDSs defined for the UK ABWR. TQUX denotes a “Transient (T) including planned manual shutdown and special initiators” followed by “failures of feedwater system (Q)”, “failures of high pressure ECCS (U)”, and “failures of reactor depressurisation (X)”.

The predicted accident progression of TQUX is as follows: Due to the large heat capacity of the suppression pool, the “high drywell pressure” setpoint for automatic depressurisation would not be reached before core damage following failure of high pressure injection. Thus, manual depressurisation by ADS or SRV followed by low pressure injection is required to prevent core damage in the Level 1 PSA. Failure of this manual depressurisation results in PDS “TQUX”.

On the other hand, the “high drywell pressure” setpoint would be reached before RPV failure because the RPV failure is estimated to be 2 to 3 hours [Ref-25.49] after the loss of high pressure injection. High D/W pressure is credited for initiating the ADS and LPFL in the Level 2 PSA except ISLOCA and BOC.

Success or failure of ADS significantly impacts the following accident progression. Therefore, the possibility of cooling the damaged core by the low pressure ECCS is considered when automatic depressurisation is successful after core damage.

If cooling of the damaged core inside the RPV by automatic depressurisation is successful, the accident is stabilised by long term heat removal from the PCV with the RHR or containment venting. If the long term heat removal fails, failure of PCV might occur.

If cooling of the damaged core inside the RPV fails after depressurisation, the accident sequence is similar to the initial conditions of PDS TQUV. After the RPV failure, the low pressure ECCS might operate to provide debris cooling and PCV spray. Even in the event of low pressure ECCS failure, the possibility of alternative water injection as part of the Accident Management (AM) measures is considered in the event tree. At the same time, the possibility of Fuel Coolant Interaction (FCI) is examined for cases with successful debris cooling. If debris cooling and heat removal are successful without occurrence of a large FCI, containment integrity would be maintained.

If the ADS do not work, the RPV is estimated to fail at high pressure in 2 to 3 hours. In that case, there is a possibility of early PCV failure due to Direct Containment Heating (DCH) and direct debris interaction with the hatch of the access tunnel and W/W to D/W vacuum breakers.

Event Tree Headings

All the headings included in the CET for TQUX are shown below. The headings included in the other CETs are described in Section 6 of the Topic Report on internal events at power Level 2 PSA [Ref-25.10].

(a) Heading “PDS TQUX”: This top heading denotes the PDS chosen.

(b) Heading “PCV Inerted prior to Initiating Event”: This heading denotes the existence or non-existence of an inerted containment at the point of the initiating event. The ABWR containment is inert with nitrogen during plant operation; however, the containment is not inert during startup or shutdown. Although startup or shutdown states account for less than 0.3 percent of power operation, this heading is considered.

(c) Heading "PCV Isolation": This heading denotes the condition of PCV isolation. Isolation valves are closed by an isolation initiation signal.

(d) Heading "RPV Depressurisation": This heading denotes possible success of RPV depressurisation after core damage, either by automatic depressurisation system (ADS), Manual Depressurisation (MDEP) or Diverse ADS (D-ADS): an alias of Reactor Depressurisation Control Facility (RDCF).

(e) Heading "Low Pressure Injection": This heading denotes possible success of injection to the RPV by low pressure injection. Low pressure injection may be achieved by the Low Pressure Core Flooder system (LPFL) and Flooder system of Specific Safety facility (FLSS). When RPV depressurisation in the preceding heading has not succeeded, this heading is failed.

(f) Heading "RPV/PCV does not fail due to FCI (In-Vessel)": This heading denotes occurrence or non-occurrence of RPV/PCV failure caused by in-vessel steam explosion due to FCI, which might occur when corium falls into the RPV lower plenum pool. Although unlikely because of the densely-packed control rod guide tubes in the lower plenum that limit energetic concerns, this heading is conservatively considered in the event that low pressure injection in the previous heading is failed.

(g) Heading "RPV intact": This heading denotes the status of the RPV - intact or failed. The probability depends on the availability of vessel injection. In the UK ABWR model, this heading succeeds if the water injection succeeds in preceding nodes related to injection.

(h) Heading "FLSS Inj into LDW": This heading denotes success or failure of the FLSS prior to failure of RPV, which acts as an alternative injection system credited for cooling the lower drywell. The UK ABWR Accident Management procedures direct injection into the lower drywell when lower RPV surface temperature reaches 300 ^{o}C. In other words, FLSS injection into LDW is conducted if the RPV injection fails. In the UK ABWR model, this heading is evaluated if there is a failure of low pressure injection before RPV failure.

(i) Heading "Lwr DW Flooder Actuates": This heading denotes success or failure of the Lower Drywell Flooder System (LDF), which acts as the passive coolant supply system credited after RPV breach. In the UK ABWR model, this heading is evaluated if there is a failure of FLSS Inj into LDW.

(j) Heading "PCV Injection Low Pressure LDW": This heading denotes success or failure of injection to the PCV by the LPFL via the RPV breach location or activation of alternate PCV injection after RPV breach. The credited alternate PCV injection system in TQUX is the mobile system – Flooding System of Reactor Building (FLSR). In the UK ABWR model, this heading is evaluated if there are failures of FLSS Injection into LDW and the Lower DW Flooder.

(k) Heading "PCV Early Fail": This heading denotes occurrence or non-occurrence of PCV failure due to FCI/ex-vessel steam explosion, DCH or Direct Debris Interaction after RPV breach. The steam explosion is potentially caused by a very large FCI when the RPV fails at low pressure and corium falls into a pool of water or if water is added on top of a hot dry pool of corium. DCH potentially occurs in high pressure melt ejection scenario if the RPV fails at high pressure. Direct Debris Interaction with the hatch of the access tunnel and W/W to D/W vacuum breakers potentially occurs in high pressure melt ejection scenario if the RPV fails at high pressure. The claimed branch probability is confirmed using an analytical approach.

(l) Heading “PCV Spray Low Pressure UDW”: This heading denotes success or failure to activate PCV spray via RHR or FLSS piping regardless of RPV breach. The spray location is upper D/W via PCV spray nozzle. The RHR pump, FLSS, and FLSR can be credited in TQUX. It is conservatively not credited that water spills over from S/P into lower drywell in case of the success of upper drywell spray because it depends on the timing of the operator action for PCV spray and accident progression.

(m) Heading “Debris Cooling”: This heading denotes occurrence or non-occurrence of PCV failure due to pedestal and basemat erosion/overpressure by MCCI. The branch probability depends on the existence of water in the lower DW at RPV failure. If the corium falls into a pool of water, it forms particulates which provide a larger surface area for debris cooling. If the corium falls into a dry lower DW, it forms a continuous pool of corium which must be cooled from the above. The claimed branch probability is confirmed using an analytical approach. Details of the analytical approach are documented in the SAA report [Ref-25.49]. If there is no water to cool the corium due to failure of injection and the LDF, debris cooling is assumed failed.

(n) Heading “Long Term Heat Removal”: This heading denotes success or failure to activate PCV heat removal function by RHR or RPV injection by LPFL via the heat exchanger (Hx). After RPV failure, temperature at PCV top head flange increases due to heat transfer from RPV even if alternative PCV spray is successful, and thus LPFL via the Hx is credited for PCV heat removal function.

(o) Heading “PCV Filtered Venting”: This heading denotes success or failure to activate containment venting function through the Filtered Containment Venting System (FCVS) or COPS. In both cases, the wetwell airspace is vented through the PCV filter.

(p) Heading “Hardened Containment Vent”: This heading denotes success or failure to activate hardened containment venting from the wetwell airspace.

(q) Heading “Failure mode”: This heading is used to distinguish two possible containment failure modes which are overpressure /overtemperature or loss of RVP support function due to MCCI erosion. The failure mode depends on the condition of LDW injection and UDW PCV spray availability as shown in the table below .

LDW injection	UDW PCV spray	Failure Mode	Notes
Failure (failure of debris cooling)	Failure	CCI	Though OP/OT (see End state “OP/OT”) is present, CCI is applied considering the additional fission product release by MCCI.
Success (failure of debris cooling)	Failure	OP/OT	OP/OT is applied because OP/OT is present and the additional fission product release by MCCI is not considered as large amount due to success of LDW injection.
Failure (failure of debris cooling)	Success	OP/OT	OP/OT is applied because OP/OT is present and the additional fission product release by MCCI is not considered to be large due to success of UDW PCV spray.
Success (failure of debris cooling)	Success	OP/OT	OP/OT is applied because the additional fission product release by MCCI is not considered to be large due to success of LDW injection and UDW PCV spray.

Event End States

The end states used for the TQUX CET are summarised below. The end states used for the other CETs are described in Section 6 of the Topic Report on internal events at power Level 2 PSA [Ref-25.10].

(a) End state “CCI”: Containment failure occurs as a result of MCCI.
Following core melt and RPV failure, pedestal and basemat erosion by MCCI also occurs if the molten core (debris) cooling in the lower D/W does not succeed. Containment failure occurs as a result of continued MCCI. For UK ABWR, fission product release occurs due to the overpressure and/or overtemperature from the non-condensable gas generated from MCCI which contributes to containment failure.
This end state is similar to the end state “OT” and/or “OP”. The end state depends on upper D/W PCV spray availability, and it is branched at the final heading “Failure mode”.

(b) End state “DCH”: Containment failure occurs by Direct Containment Heating (DCH) in this end state.
DCH occurs when high pressure RPV is breached, e.g. the case without RPV depressurisation (under high pressure conditions). Because of this, the PCV is damaged due to highly energetic events associated with RPV failure. DCH under low RPV pressure conditions is not considered in this PSA.

(c) End state “DI”: Containment failure occurs by direct debris interaction in this end state.
Direct debris interaction occurs when high pressure RPV is breached, e.g. the case without RPV depressurisation (under high pressure conditions). Because of this, the debris directly interacts with the hatch at lower drywell tunnel and/or W/W to D/W vacuum breakers. Direct debris interaction under low RPV pressure conditions is not considered in this Level 2 PSA.

(d) End state “FVP”: Although PCV venting occurs, containment failure does not occur in this end state. Although core melt and RPV failure occur, molten core (debris) is cooled in the lower D/W by PCV injection or PCV spray. However, long term heat removal does not succeed. Because of this, an increase of PCV pressure is prevented by PCV filtered venting.

(e) End state “FVV”: Although PCV venting occurs, containment failure does not occur in this end state. Although RPV failure does not occur, long term heat removal does not succeed. Because of this, an increase of PCV pressure is prevented by PCV filtered venting.

(f) End state “KP”: Containment is intact in this end state. Although core melt and RPV failure occur, molten core (debris) is cooled in the lower D/W by PCV injection or PCV spray, resulting in successful long term heat removal.

(g) End state “KV”: Containment is intact in this end state. In this end state, the RPV remains intact and long term heat removal is successful.

(h) End state “OP/OT”: Containment failure occurs due to combination of overpressure and overtemperature. The failure location is assumed at the drywell head. The criterion to ensure PCV intact is described in section 25.5.2.3. This end state occurs when long term heat removal, PCV venting and PCV spray are unsuccessful.

(i) End state “OP/OT_PS”: Containment failure occurs due to combination of overpressure and overtemperature. The failure location is assumed at the drywell head. The criterion to ensure PCV intact is described in section 25.5.2.3. However, in this case PCV spray to the upper drywell is successful.

(j) End state “PCI”: This end state does not assume containment failure, but the frequency of this end state is considered for Large Release Frequency (LRF) because fission products are released by PCV isolation failure. Even when the PCV isolation failure occurs, fission product release can be reduced by mitigation functions.

(k) End state “PE”: Containment failure occurs by ex-vessel steam explosion due to Fuel Coolant Interaction (FCI). FCI might occur either when molten debris falls into water or when water is injected to molten debris. The impulse by steam explosion is considered.

(l) End state “VP”: Although PCV venting occurs, containment failure does not occur in this end state. Although core melt and RPV failure occur, molten core (debris) is cooled in the lower D/W by PCV injection or PCV spray. However, long term heat removal does not succeed. Because of this, an increase of PCV pressure is prevented by PCV venting through the hardened containment vent.

(m) End state “VV”: Although PCV venting occurs, containment failure does not occur in this end state. Although RPV failure does not occur, long term heat removal does not succeed. Because of this, an increase of PCV pressure is prevented by PCV venting through the hardened containment vent.

(n) End state “RE”: Containment failure occurs by in-vessel steam explosion due to FCI. FCI might occur when corium falls into the pool of RPV lower plenum. The impulse by steam explosion is considered.

(o) End state “BYPASS”: Containment bypass event resulting in direct radiation release. This end state is modelled as a direct connection to the RPV to the environment.

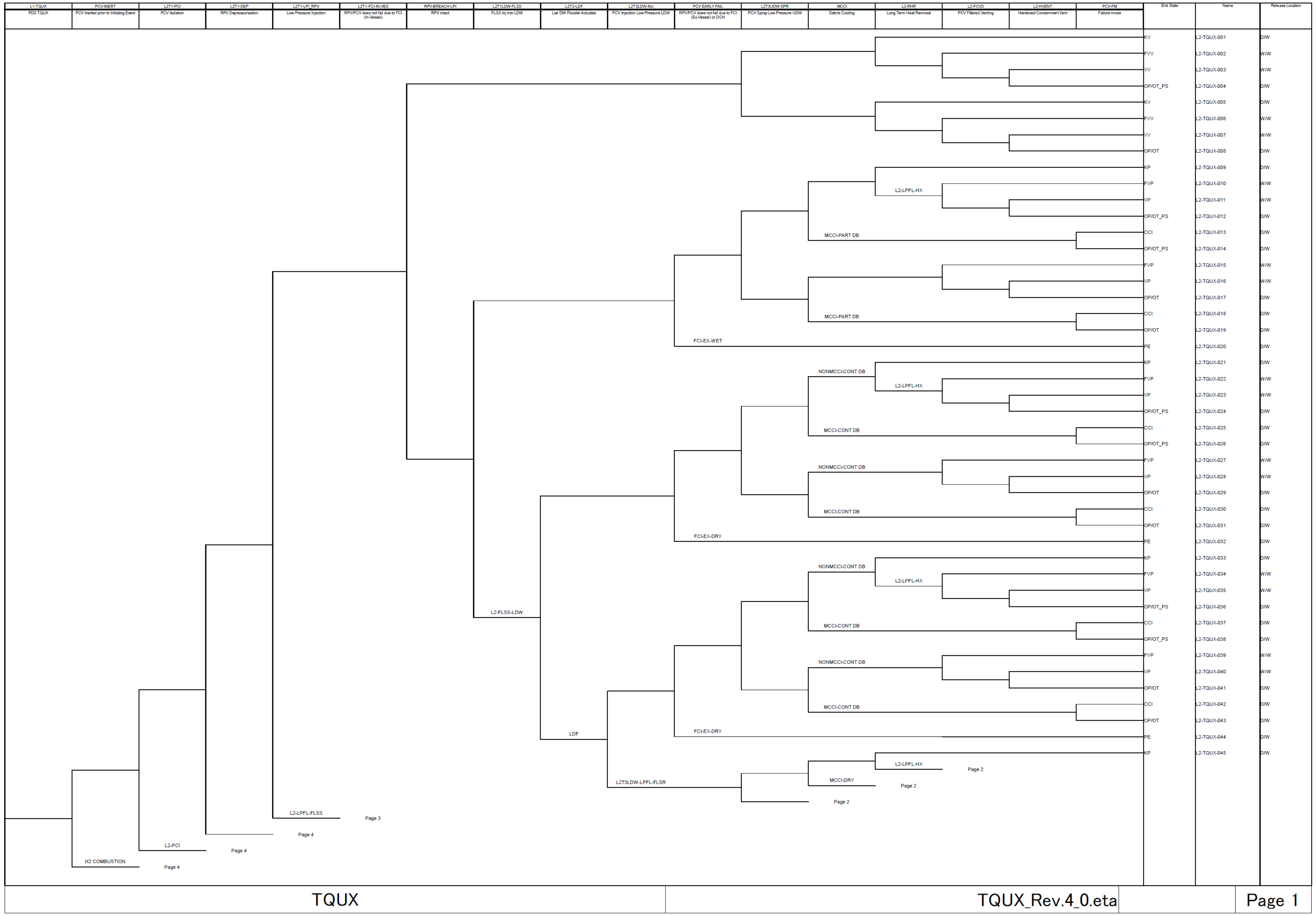

Figure 25.5.2-2 Containment Event Tree of TQUX (1/4)

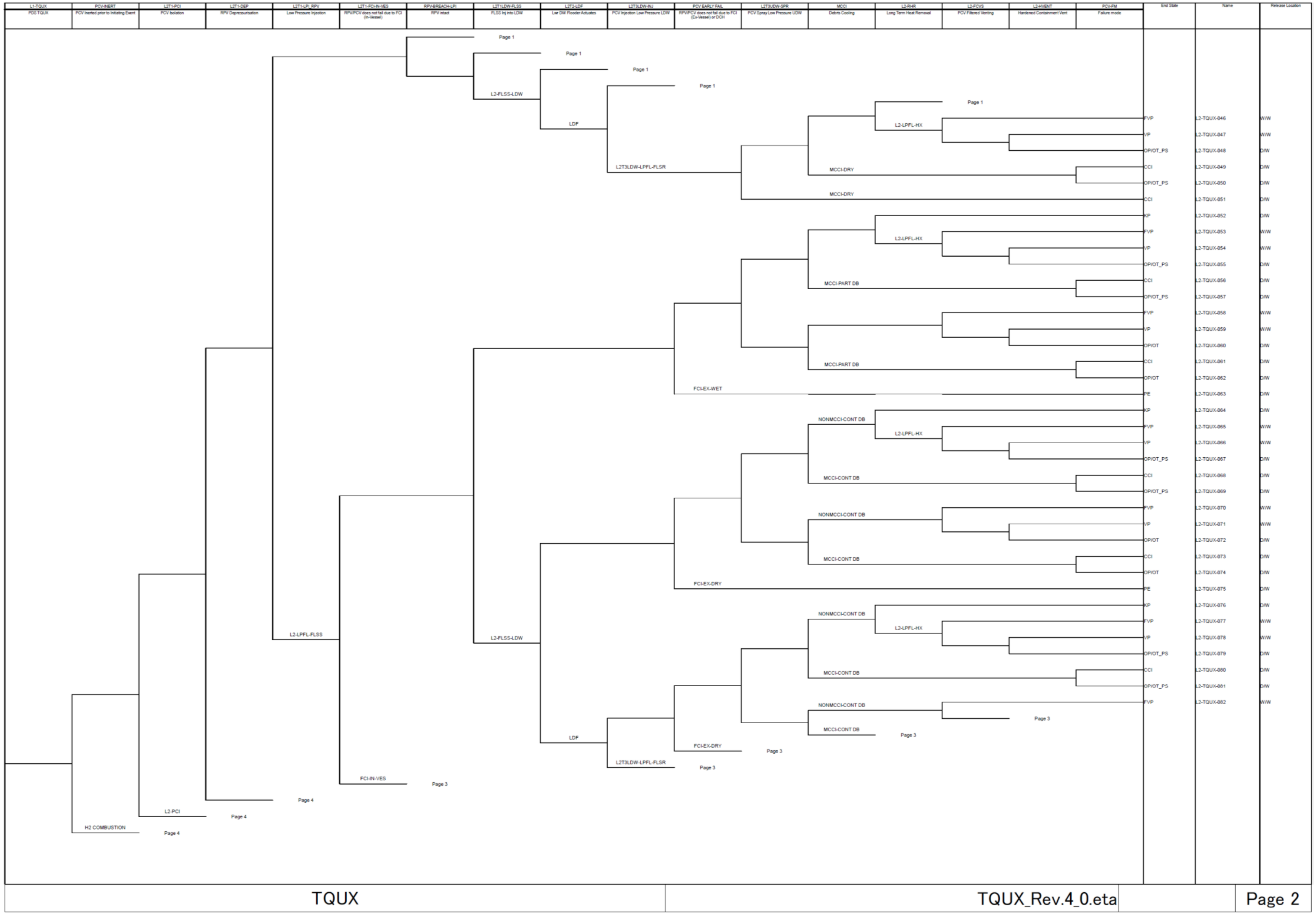

Figure 25.5.2-2 Containment Event Tree of TQUX (2/4)

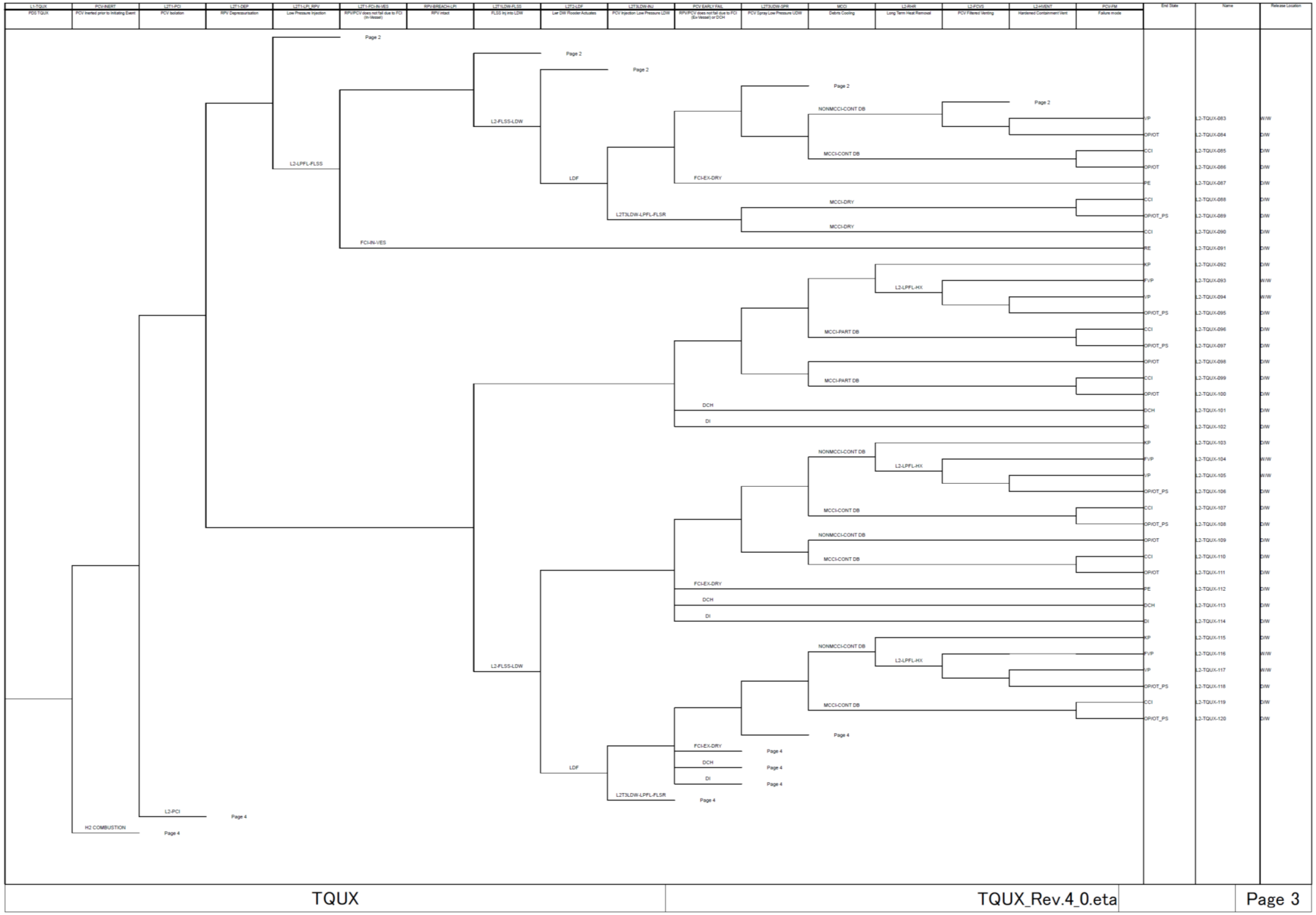

Figure 25.5.2-2 Containment Event Tree of TQUX (3/4)

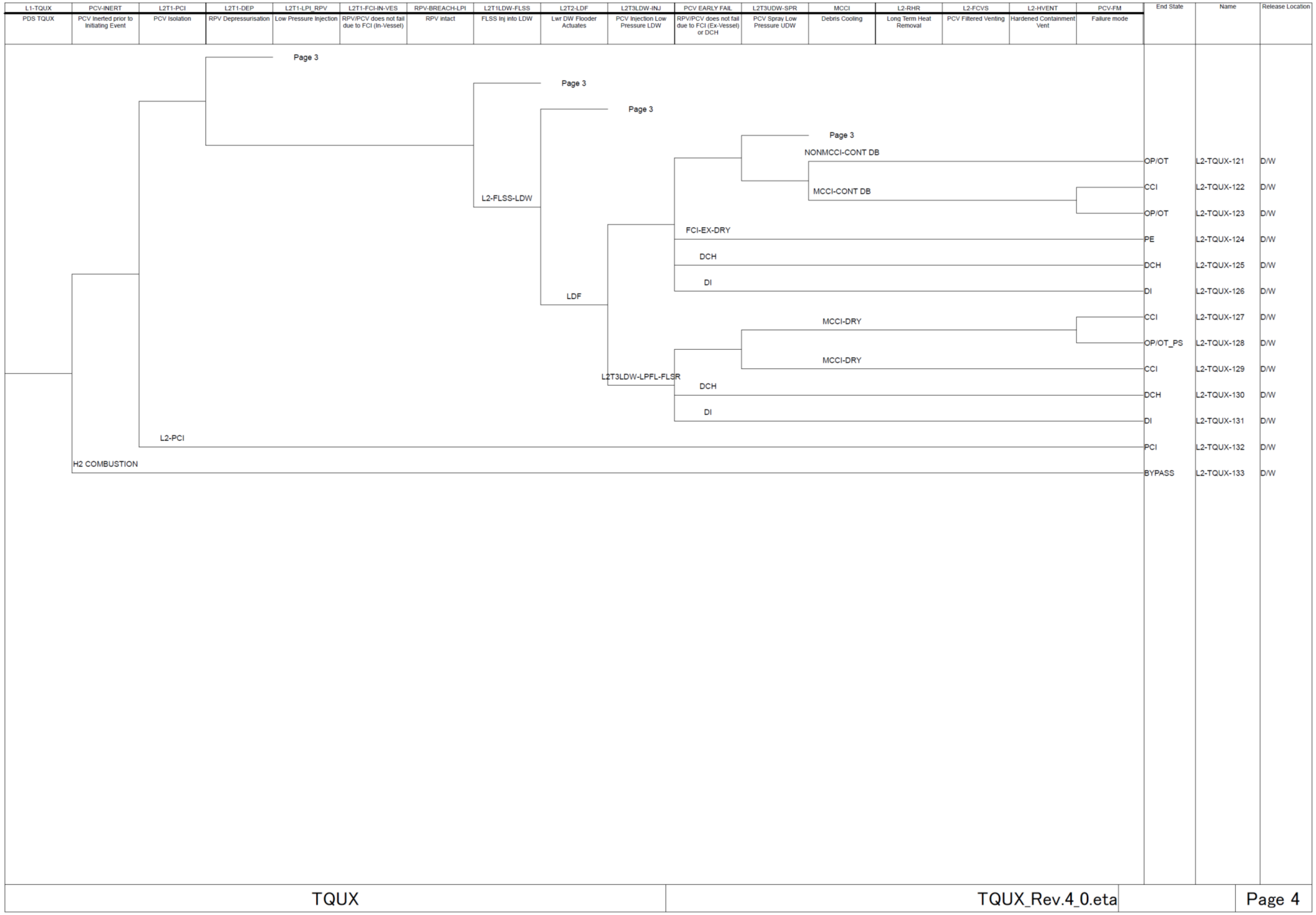

Figure 25.5.2-2 Containment Event Tree of TQUX (4/4)

(2) Branch Probability for Phenomenological Event

Based on the approach summarised in Section 25.5.2.1(1) for the TQUX PDS, branch probabilities for phenomenological events were evaluated based on NUREG/CR-4700 [Ref-25.50] engineering judgement in Table 25.5.2-2.The containment failure modes considered are described below.

- In-Vessel FCI: Containment failure due to RPV top head missile by in-vessel steam explosion
- Ex-Vessel FCI: Containment failure due to pedestal failure by Ex-vessel steam explosion
- DCH: Containment failure by Direct Containment Heating
- MCCI: Containment failure (e.g. overtemperature, overpressure, or excessive pedestal erosion) by long term MCCI
- Direct Debris Interaction: Containment failure due to debris direct interaction with access hatch and W/W to D/W vacuum breakers

Table 25.5.2-3 shows the branch probabilities defined for these phenomenological events.

In-Vessel FCI

In-vessel FCI might occur when the corium falls into the RPV lower plenum pool (the resulting PCV failure mode is called α-mode). A value of 1E-4 (Impossible) was applied to branch probabilities for this phenomenon. The branch probability was based on NUREG/CR-4700 [Ref-25.50] engineering judgement in Table 25.5.2-2 and it is considered conservative because in-vessel steam explosions leading to containment failure are not credible events as discussed in NUREG/CR-5960 [Ref-25.56]. NUREG/CR-5960 states that because of the densely-packed control rod guide tubes in the lower plenum, BWRs are not subject to in-vessel steam explosion energetic concerns. However, in-vessel FCI is explicitly modelled in the CETs to represent the phenomena and accident sequence.

Ex-Vessel FCI

Containment failure with FCI immediately after vessel breach is considered for all PDSs. Values of 1E-2 (Very unlikely) with pedestal water filling and 1E-3 (Highly Unlikely) without pedestal water filling were applied to branch probabilities. These branch probabilities were based on NUREG/CR-4700 [Ref-25.50] engineering judgement in Table 25.5.2-2 and it is considered conservative as is usually claimed that FCI only occurs with debris dropped into the pedestal pond.

DCH

Containment failure resulting from DCH immediately after vessel breach is considered in PDSs with RPV high pressure breach. A value of 1E-2 (Very Unlikely) with the RPV high pressure breach was applied to this branch probability. For the case of low pressure RPV failure, it was judged that containment failure by DCH is “practicably negligible”. This branch probability was based on NUREG/CR-4700 [Ref-25.50] engineering judgement in Table 25.5.2-2 and it is considered conservative because DCH only occurs in a High Pressure Melt Ejection (HPME) scenario.

MCCI

Containment failure with MCCI is considered in all PDSs with vessel breach sequence. Values of 1E-2 (Very Unlikely) with pedestal water filling before vessel breach, 1E-1 (Unlikely) with water filling after vessel breach and 1.0 (Certain) without water filling were applied to branch probability. This branch probability was based on NUREG/CR-4700 [Ref-25.50] engineering judgement in Table 25.5.2-2 and it is considered conservative or comparable according to analytical approach.

Direct Debris Interaction

Containment failure with Direct Debris Interaction instantaneously after vessel breach is considered in PDSs with RPV high pressure breach. A value of 1.0 with the RPV high pressure breach was conservatively applied to branch probability.

Considering radiation heating from 1,000 K debris on the lower D/W structures to the access tunnel and V/B structures as the worst case, the access tunnel and the V/Bs are damaged by overtemperature.

Conclusion

It was confirmed that phenomenological branch probabilities based on NUREG/CR-4700 [Ref-25.50] were conservative or comparable to the analytical approach as described in Section 25.5.2 of [Ref-1]. In general phenomenological probability has a large uncertainty. Therefore, conservative probabilities based on NUREG/CR-4700 have been used for this PSA while sensitivity analysis has been performed utilising the probabilities evaluated by analytical approach instead of the branch probabilities based on NUREG/CR-4700.

Table 25.5.2-2 Branch Probabilities Used Based on NUREG/CR-4700 [Ref-25.50]

Category	Probability Area	Branch probability
Certain	Probability=1	1
Highly Likely	1.0> Probability>0.995	0.999
Very Likely	0.995> Probability>0.95	0.99
Likely	0.95> Probability>0.7	0.9
Indeterminate	0.7> Probability>0.3	0.5, 0.3
Unlikely	0.3> Probability>0.05	0.1
Very Unlikely	0.05> Probability>0.005	0.01
Highly Unlikely	0.005> Probability>0.0	0.001
Impossible	Probability=0.0	0.0001

Table 25.5.2-3 Branch Probability for Phenomenological Event

Phenomenological event	Mean	Beta distribution (alpha parameter)*	Basic Event
In-Vessel FCI	1.0E-4	0.3	FCI-IN-VES
Ex-Vessel FCI (with pedestal water filling)	1.0E-2	0.3	FCI-EX-WET
Ex-Vessel FCI (without pedestal water filling)	1.0E-3	0.3	FCI-EX-DRY
DCH	1.0E-2	0.3	DCH
MCCI (without water filling)	1.0	-	MCCI-DRY
MCCI (without water filling prior to RPV breach)	1.0E-1	1	MCCI-CONT_DB
MCCI (with water filling prior to RPV breach)	1.0E-2	0.3	MCCI-PART_DB
Direct Debris Interaction	1.0	-	DI

*Alpha parameter is estimated to match the upper and lower bound value as shown in Table 25.5.2-2. If it is not possible to match both, the upper bound value is utilised rather than the lower bound value.

(3) Operator Actions and Human Reliability Analysis (HRA)

Even if the plant is in a core damage condition, operator actions are required to reduce the fission product release. In this section, the treatment of operator actions and HRA in the Level 2 PSA are described.

HRA for the UK ABWR Level 2 PSA was assessed using the same methodology as that for Level 1 PSA.

[Chapter 27 of PCSR] describes the technical work process related to Human Factors including HRA.

HRA for the PSA on the pre-initiator human failure events (Type A HFEs) have been performed using the following process.

- Identification of potential human failure events that result in loss of function of mitigation systems
- Qualitative task analysis
- HEP quantification

HRA for the PSA on the post-initiator human failure events (Type C HFEs) have been performed in the following process.

- Determination of post-initiator operator actions credited in the PSA
- Qualitative task analysis
- HEP quantification

Time windows have been developed for HRA. The time windows for the post-initiator operator actions were defined in the assessment. FLSR and DAG were credited for the accident sequences which have a time window of at least 8 hours for activation of these systems. A summary of calculated HEPs for post-initiator HFEs in Level 2 PSA is shown in Table 25.5.2-4.

In the Level 2 PSA, two kinds of inter-HFE dependencies were considered. One was the dependency between the Level 1 PSA and the Level 2 PSA. The other was the dependency between HFEs modelled in the Level 2 PSA. The former dependency was provided by the conditional probability given the failure of the operator action before core damage as shown in Table 25.5.2-5. The latter dependency was provided by two kinds of common “cognition failure” as shown in No. L2-12 and L2-13 of Table 25.5.2-5.

The adequacy of dependency in each combination of cutsets related to HFEs was reviewed to identify inter-HFE dependency.

Table 25.5.2-4 Summary of Calculated HEPs for Post-initiator HFEs in Level 2 PSA [Ref-25.10]

HFE No.	Basic event ID	Description
L2-1	___-HFE-DP-L2	Failure of manual RPV depressurisation during transient (Back-up of transient ADS) for Level 2 PSA
L2-2	___-HFE-FS-L2	Failure of depressurisation and FLSS initiation for Level 2 PSA
L2-3	___-HFE-FL-L2	Failure of lower drywell injection with FLSS
L2-4-1	___-HFE-FU-L2	Failure of upper drywell spray with FLSS
L2-4-2	___-HFE-FU-2S	Failure of upper drywell spray with FLSS during SBO
L2-5	___-HFE-RH-L2	Failure of manual initiation of RHR for drywell spray and long term heat removal
L2-6-1	___-HFE-CV-L2	Failure of containment venting to prevent PCV overpressure (generic)
L2-6-2	___-HFE-CV-2S	Failure of containment venting to prevent PCV overpressure during SBO
L2-7	___-HFE-IS-L2	Failure of containment isolation to prevent release of fission products
L2-8	___-HFE-LP-L2	Failure of lower drywell injection via RPV with LPFL
L2-9	FLSR_UNAVAILABLE*	Failure of lower drywell injection with FLSR
L2-10	FLSR_UNAVAILABLE*	Failure of upper drywell spray with FLSR
L2-11	DAG*1	Failure of AC power supply with DAG
L2-12	___-HFE-CC-L2	Failure of cognition for damaged core cooling (HEPs before RPV breach (No. L2-1 and L2-2) have the common “cognition failure”.)
L2-13	___-HFE-DC-L2	Failure of cognition for debris cooling and containment heat removal (HEPs after RPV breach (No. L2-3 to L2-11) have the common “cognition failure” except L2-7.)

*This basic event includes the component failure, T&M unavailability and HEP.

Table 25.5.2-5 Treatment of Inter-HFEs Dependencies

No.	Basic event ID	Description	Treatment of dependencies
1	__-HFE-MD-TR (modelled in Level 1 PSA)	Failure of manual RPV depressurisation during transient (Back-up of transient ADS)	The conditional probability given the failure of operator action before core damage is applied to "___-HFE-DP-L2".
	___-HFE-DP-L2	Failure of manual RPV depressurisation during transient (Back-up of transient ADS) for Level 2 PSA	
2	___-HFE-FL-IN (modelled in Level 1 PSA)	Failure of depressurisation and FLSS initiation after failure of ECCS	The conditional probability given the failure of operator action before core damage is applied to "___-HFE-FS-L2".
	___-HFE-FS-L2	Failure of depressurisation and FLSS initiation for Level 2 PSA	
3	___-HFE-DP-L2	Failure of manual RPV depressurisation during transient (Back-up of transient ADS) for Level 2 PSA	HEPs before RPV breach (Core cooling strategy) have the common cognition failure "___-HFE-CC-L2" to represent the dependency.
	___-HFE-FS-L2	Failure of depressurisation and FLSS initiation for Level 2 PSA	
4	___-HFE-FL-L2	Failure of lower drywell injection with FLSS	HEPs after RPV breach (containment control strategy) have the common cognition failure "___-HFE-DC-L2" to represent the dependency.
	___-HFE-FU-L2	Failure of upper drywell spray with FLSS	
	___-HFE-RH-L2	Failure of manual initiation of RHR for drywell spray and long term heat removal	
	___-HFE-CV-L2	Failure of containment venting to prevent PCV overpressure	

25.5.2.2 Severe Accident Analysis

(1) Purpose of Severe Accident Progression Analysis

Accident progression analyses of each PDS were performed to obtain the data necessary for the development of the CETs, including plant hydrological behaviour, chronology of accident progression (core damage, containment failure), and containment load from severe accident phenomena [Chapter 26 of PCSR].

(2) Evaluation Method for Severe Accident Progression Analysis

The MAAP code (version 4.07) was used for accident progression analysis to support the Level 2 PSA for the UK ABWR. This analysis included models for the significant accident phenomena that might occur in the RPV, in the containment, and in the reactor building. The MAAP code calculated the progression of the postulated accident sequence, including the deposition of the fission products, from initiating events to either a safe, stable state or to an impaired containment condition (due to overpressure or overtemperature). The MAAP code also predicted fission product releases to the environment. The applicability of the MAAP code for modelling the UK ABWR is described in [Chapter 26 of PCSR].

(3) Representative Sequences for Severe Accident Progression Analysis

As described in section 25.5.1.2, Accident Classes were categorised into twenty-six sequences, which were taken over as PDS in the Level 2 PSA.

In the PDSs, the following sequences demonstrate very similar accident progression as other PDS:

- TBU and TBD follow the same accident progression as TQUX although the availability of AC/DC power is different.
- TBP follows the same accident progression as TQUV although the availability of AC power is different.
- TBPN follows the same accident progression as TNQUV although the availability of AC power is different.
- TBDN follows the same accident progression as TNQUX although the availability of DC power is different.

Therefore, twenty-one (= 26-5) PDSs were selected as the unmitigated analyses cases [Ref-25.49].

The selection of initiating events and the modelling of the loss of functions were based on the characteristics of plant responses and effect on fission product releases as follows:

- Loss of feedwater flow was selected as the initiating event in the sequences of TQUV, TQUX, TW, TW-LP, TC-HP, TC-LP, TNQUV and TNQUX. It leads to earlier core damage and earlier RPV failure.
- Closure of MSIVs was selected as initiating event in the sequences of TC and TCN. It leads to earlier PCV failure and earlier core damage.
- Break/isolation failure of RHR line was selected as initiating event in the sequence of AE, AW-LP, AC, S3E and AC. As the RHR line is the largest piping in the lowest place of RPV when RPV is depressurised due to LOCA. It leads to earlier core damage and earlier RPV failure.

- Break/isolation failure of bottom drain line is selected as initiating event in the sequence of S3UX, S12UX, S12W and S21C. As the bottom drain line is the largest piping in the lowest place of RPV when RPV is not depressurised due to LOCA. It leads to earlier core damage and earlier RPV failure.

(4) Accident Progression Analysis Conditions

The analysis conditions assumed for all accident sequences are indicated in Table 25.5.2-6. The analysis conditions are applied to both the unmitigated analysis and the mitigated analysis. The ultimate capacity of the ABWR RCCV is more than 200 °C and at least 620 kPa [gauge][Ref-25.49].

Although the UK ABWR has measures to prevent the occurrence and propagation of the severe accident, none of these measures were considered in the unmitigated analysis because these analyses are conducted to obtain the following key parameters for the PSA.

- Timing of core support plate failure - used to provide the time margin for RPV depressurisation and water injection into RPV.
- Timing of RPV failure - used to provide the time margin for lower drywell injection for pre-flooding.
- Timing of PCV failure due to overpressure and overtemperature - used to provide the time margin for RHR initiation, RHR recovery, offsite power recovery, DG recovery, AC bus recovery, DC bus recovery and PCV venting.
- Timing of pedestal failure - used to provide the time margin of lower drywell injection for post-flooding.

(5) Accident Progression Analysis Results for Unmitigated Sequences

The timings of key events for each unmitigated accident sequence were calculated using the MAAP code (version 4.07) [Chapter 26 of PCSR]. Table 25.5.2-7 shows the analysis results.

(6) Accident Progression Analysis on Mitigated Sequences

The timings of key events for each mitigated severe accident sequence, and success criteria for the Level 2 PSA were also calculated using the MAAP code (version 4.07) [Chapter 26 of PCSR]. A summary of success criteria for mitigation system operation that was developed based on the analysis is shown in Table 25.5.2-8.

Table 25.5.2-6 Analysis Conditions of Severe Accident Progression Analysis [Ref-25.49]

Items	Conditions	Remarks
Initial reactor power	3,926 MWt	Rated power
Initial core flow rate	52.2×10^3 t/h	Rated flow rate
Initial reactor pressure	7.07 MPa [gauge]	Rated pressure
Initial reactor water level	Normal level	Normal water level
Free volume of PCV	D/W Open Space: 7,350 m^3 W/W Open Space: 5,960 m^3	Design value
Water inventory of S/P	3,580 m^3	Design value
Initial water temperature of S/P	35 ^oC	Maximum temperature during normal operation
Ultimate pressure of PCV	Pressure: 620 kPa [gauge]	Twice maximum design pressure of PCV
Ultimate temperature of PCV	Temperature: 200 ^oC	Judged by PCV gas temperature
Decay heat power	ANSI/ANS-5.1-1979	MAAP model
Fuel type	GE-14	10x10 fuel rods
Number of core nodes	Radial Nodes: 5 rings Axial Nodes: 28 nodes	MAAP standard setting
Leakage area of PCV	0 m^2	No leakage is assumed to evaluate pressure and temperature conservatively.
Failure area of PCV	0.068 m^2	Large failure area due to overpressure and overtemperature is assumed.

Table 25.5.2-7 Summary of Accident Progression Analysis Results

Case	PDS	Event Timing (hr)						RPV failure mode	Remarks
		Core damage	Core support plate failure	RPV failure	PCV pressure reaches 1 Pd	PCV failure	Pedestal failure		
1	S4	0.0	1.2	0.0	0.0	0.0	21.5	RPV lower head failure	
2	AE	0.2	2.0	6.7	7.3	7.4	over 20	*	
3	TQUV	0.6	3.1	6.8	9.9	16.6	over 20	*	
4	TQUX	0.6	3.1	4.8	8.1	13.5	over 20	*	
5	TB	11.8	16.5	18.8	16.5	18.8	over 50	*	
	TBU	0.6	3.1	4.8	8.1	13.5	over 20	*	Same as TQUX
	TBP	0.6	3.1	6.8	9.9	16.6	over 50	*	Same as TQUV
	TBD	0.6	3.1	4.8	8.1	13.5	over 20	*	Same as TQUX
6	TW	25.1	31.1	33.2	15.8	21.5	over 50	*	
7	TW-LP	21.9	26.7	36.1	11.1	16.6	over 50	*	
8	AW-LP	18.9	23.3	32.6	7.9	15.2	over 50	*	
9	AC	1.5	3.7	9.2	0.7	1.0	29.1	*	
10	TC	1.2	4.0	5.7	0.6	1.0	over 30	*	
11	TC-HP	0.2	2.1	3.8	7.0	12.3	over 20	*	
12	TC-LP	0.2	2.1	5.8	10.0	16.5	over 20	*	
13	S3E	0.2	1.8	6.9	-	-	25.8	*	
14	S3UX	0.5	2.6	4.1	8.4	14.3	20.7	*	
15	TNQUV	0.6	3.2	7.0	0.3	0.8	over 20	*	
16	TNQUX	0.6	3.0	4.7	0.3	0.8	-	*	
	TBPN	0.6	3.2	7.0	0.3	0.8	over 20	*	Same as TNQUV
	TBDN	0.6	3.0	4.7	0.3	0.8	-	*	Same as TNQUX
17	TCN	0.2	2.1	3.9	0.0	0.0	over 30	*	
18	AN	2.6	5.3	11.7	-	0.3	-	*	
19	S12UX	0.4	2.2	4.0	7.7	12.7	22.5	*	
20	S12W	19.9	23.8	27.1	12.3	19.0	-	*	
21	S12C	2.1	4.4	6.8	1.5	2.0	39.1	*	

*EJECTION OF CRD TUBES

Table 25.5.2-8 Summary of Success Criteria for UK ABWR Level 2 PSA

No.	Success criteria	Timing	Mission time [g]
RPV injection to achieve in-vessel recovery			
1	1 SRV [a] + 1/3 LPFL [b] (Manual or Automatic)	After receiving automatic initiation signal (L1 or drywell pressure high) Before core support plate fails	24 h
2	1 SRV [c] + 1/4 FLSS [b] (Manual or Automatic)	After receiving automatic initiation signal (L1) Before core support plate fails	24 h
Lower drywell injection			
3	1/4 FLSS (Manual) [d]	After lower head temperature reaches 300 ^{o}C	24 h
4	2/10 LDF (Passive) [e]	After gas temperature of lower drywell reaches 260 ^{o}C	24 h
5	1/1 FLSR (Manual) [d]	After debris falls into lower drywell	24 h
6	1/3 LPFL via RPV breach location (Manual)[d]	After debris falls into lower drywell	24 h
Upper drywell spray			
7	1/4 FLSS (Manual) [f]	After D/W pressure or temperature Before D/W pressure reaches 1.5 Pd or D/W temperature reaches 200 ^{o}C	24 h
8	1/1 FLSR (Manual) [f]	After D/W pressure or temperature Before D/W pressure reaches 1.5 Pd or D/W temperature reaches 200 ^{o}C	24 h
9	1/3 RHR (Manual) [f]	After D/W pressure or temperature Before D/W pressure reaches 1.5 Pd or D/W temperature reaches 200 ^{o}C	36 h
Long term heat removal			
10	1/3 RHR (Automatic or Manual)	<u>RPV intact:</u> After receiving automatic initiation signal (S/P temperature high for SPC mode) <u>RPV failure:</u> LPFL mode via heat exchanger Before D/W pressure reaches 2.0 Pd or D/W temperature reaches 200 ^{o}C	24 h
11	Manual containment venting or COPS (Passive)	After containment design pressure, or immediately after reaching COPS setpoint (twice of containment design pressure)	24 h

a: 3 SRVs are conservatively assumed in the fault tree for the Level 2 PSA. The CCF probability of SRVs is calculated by hand in the Level 1 PSA. The CCF is also modelled in the Level 2 PSA to consider the dependency between Level 1 and Level 2 PSA.
b: The capacity of one pump is larger than 90 m^3/h, which is the required injection rate.
c: 6 SRVs are conservatively assumed in the fault tree for the Level 2 PSA. The CCF probability of SRVs is calculated by hand in the Level 1 PSA. The CCF is also modelled in the Level 2 PSA to consider the dependency between Level 1 and Level 2 PSA.
d: The capacity of one pump is larger than 60 m^3/h, which is the required injection rate.
e: Required injection rate is 60 m^3/h and design flow rate is 40 m^3/h/line.
f: The capacity of one pump is larger than 300 m^3/h, which is the required injection rate.
g: In Level 2 PSA analyses, a typical equipment mission time is 24 hours after the onset of core damage. However, the mission time to determine the magnitude and characteristics of the fission product releases is 36 hours after the onset of core damage.

25.5.2.3 Containment Performance Analysis

The containment performance analysis consisted of three subtasks:

- Subtask 1: Identification of Containment Failure Modes,
- Subtask 2: Containment Structural Analysis, and
- Subtask 3: Analysis of Containment Bypass Event and Containment Isolation Failure.

(1) Subtask 1: Identification of Containment Failure Mode

The PCV fails when the safety features of RCCV or mitigation systems work insufficiently to control the accident progression. The containment failure modes are identified through the examination of containment capacity analysis. Among identified containment failure modes, containment failure modes as shown in Table 25.5.2-9 are explicitly modelled in the CETs. In addition, intact PCV sequences are treated in CETs as shown in Table 25.5.2-10.

The Level 2 PSA CET survey, literature survey and screening of SA phenomena, key SA phenomena are extracted in Table 25.5.2-11. Overpressure and overtemperature, in-vessel / ex-vessel Fuel Coolant Interaction, Direct Containment Heating, Molten Core Concrete Interaction, Direct Debris Interaction are key SA phenomena for further evaluation. Sub atmospheric containment pressure and hydrodynamic loading are not considered in Level 2 PSA as the evaluation results in this section.

Containment Strength for Overpressure / Overtemperature

In severe accidents, the Containment is heated up by the heat generation of the water – zirconium reaction, and the decay heat. Deterioration of containment function of the PCV is often caused by the heat damage to the PCV boundaries such as the access hatch gasket made of non-metal materials. Based on the results of historical analysis and experiments, it is known that the silicone-rubber gaskets and sealer based on organic resin in PCV can maintain capability of sealing up to at least 200 °C, 620 kPa [gauge] [Ref-25.49].

Discussion of Direct Containment Heating (DCH)

Direct Containment Heating (DCH) was added to the probabilistic safety assessment report (NUREG-1150) by the Nuclear Regulatory Commission (NRC) in the USA. The DCH is one of the phenomena which may cause the PCV failure. If the reactor vessel fails at high pressure, the corium in the lower plenum is driven into the lower D/W with high velocity by the pressure difference between the RPV and the lower D/W. This phenomenon is called High Pressure Melt Ejection (HPME). Based on our experience for severe accident evaluations, RPV failure due to CRD tube ejection is the most likely failure mode. When HPME occurs, the surface of the corium jet is torn by the high velocity gas flow, generating fine corium droplets. The fragmented corium may be released and spread into the PCV. The temperature and pressure of the gaseous atmosphere in the PCV rapidly rises due to reaction heat and hydrogen generation due to the steam oxidation reaction of Zr on the surface of the fragmented corium, and the direct heat transfer from the fragmented corium to gas in the PCV. As a result of these phenomena, pressure and temperature in the PCV rises. If the maximum pressure exceeds the ultimate pressure capacity of the PCV, it can lead to the PCV damage. This series of phenomena are called DCH.

The probability of the DCH occurring is extremely small for the UK ABWR for two reasons. First, the UK ABWR has many diverse mechanisms to depressurise the RPV. The second reason is the UK ABWR has ten connecting venting pipes which have small flow area. Although they connect the lower drywell with upper drywell and wetwell, a large part of steam flows into wetwell and the fragmented corium is cooled in the suppression pool. Therefore, DCH hardly occurs in the PCV in the UK ABWR. However, if DCH were to occur, the possibility of an early release of a large fission product would be high and there would be the possibility of high exposure to the public.

In the past, DCH has principally been a concern for PWRs. BWRs have very reliable vessel depressurisation systems. Therefore, the frequency of accidents with the vessel remaining at high pressure is extremely low. In addition, it is known that depressurisation of the Unit 1 RPV at Fukushima has been confirmed; however, a manual activation of SRVs was not conducted in the accident of Fukushima nuclear power station unit 1 [Ref-25.52]. The cause of the depressurisation is estimated to be not due to RPV failure but due to gas leakage through instrumentation tubes in the core or the gasket on the steam line because PCV temperature and PCV pressure did not increase rapidly at the timing of depressurisation. As the UK ABWR has a lot of instrumentation tubes in the core and gaskets on the steam line, failure of some instrumentation tubes is likely to occur and the leakage area produced during a severe accident. If leakage from instrumentation tubes is produced, the possibility of DCH occurrence will decrease.

Discussion of ex-vessel Fuel Coolant Interaction (FCI)

Generally, the steam explosion can be divided into the four elementary processes [Ref-25.53].

- Pre-mixing state
- Spontaneous or external trigger
- Interactional propagation accompanied with rapid heat transfer and fine fragmentation
- Release of mechanical energy due to expansion

A steam explosion occurs when all the above four conditions are achieved. In order to effectively convert the thermal energy of the high temperature corium to the mechanical energy, a large amount of the high temperature corium must effectively form a pre-mix condition in which the corium particle size can be easily atomised. Next, a trigger must break the steam film on the surface of a pre-mixed corium particle to initiate a steam explosion. The stability of the vapour film on the pre-mixed corium particle depends on the water temperature. In case of the low subcooled water, the steam film is stable because a vapour film break due to condensation rarely occurs. Therefore, a steam explosion due to the spontaneous trigger is unlikely to occur in low subcooled water. After the triggering occurs, the break of the vapour film may cause the vapour film of neighbouring corium particles to break in the same way. The propagation velocity strongly depends on the pre-mixing condition. If the pre-mixture condition is favourable, propagation velocity might increase to a detonation wave and a steam explosion might occur. A steam explosion is caused when all of these four elementary processes are satisfied. In the lower D/W, Ex-Vessel steam explosion might occur when the corium jet flows into the water pool in the lower D/W, or when the water is injected onto the upper surface of the continuous corium accumulated on the lower D/W by LDF or Accident Management.

If a severe accident progresses and the RPV is failed, the corium discharges to the outside of the RPV. If there is a water pool on the floor of the lower drywell, the pool water contacts the corium and there is a possibility of occurrence of a steam explosion. A steam explosion may induce a load that threatens the integrity of the PCV.

When the amount of water on the floor of the lower D/W is small or if the water temperature is high, a steam explosion is unlikely to occur for two reasons. The first reason is the energy transfer from the corium to the water is limited by the small amount of water. The second reason is the high vapour film stability at the surface of the corium particle because of the high temperature water which can evaporate easily. Therefore, a steam explosion with such condition does not need to be considered.

When there is a large amount of water on the floor of the lower D/W, a steam explosion does not occur easily. For ex-vessel FCI, a lot of large-scale experiments and studies have been conducted since the 1970's. In the FARO, the KROTOS and the COTELS tests, which are experiments using UO_2 corium, steam explosion did not occur except for those cases which sent shock waves to the corium from outside as an external trigger [Ref-25.53],[Ref-25.54].

In addition, even if a steam explosion occurs in the UK ABWR, the mechanical energy discharged from steam explosion does not affect the soundness of pedestal structures. The capability of the containment to withstand rapid pressurisation and impulse loads to the pedestal has been evaluated for the standard ABWR. The demonstration of pedestal capability is one of the requirements of severe accident assessment within the design of the civil aspects of containment.

Discussion of Molten Core Concrete Interaction (MCCI)

The corium that falls into the lower D/W floor generates decay heat. If corium cooling is insufficient, the corium thermally decomposes and erodes the concrete, releasing H_2O and CO_2 contained in the concrete as gasses. When H_2O and CO_2 rise through the corium due to buoyancy, they cause chemical reactions with the high temperature corium and are deoxidised to H_2 and CO, which are combustible, non-condensable gasses. Additionally, the chemical form of the fission products contained in the corium changes and there is a possibility that fission products aerosols are additionally released from the corium to the PCV. Further, if MCCI continues, the concrete continues to be eroded and there is a possibility that the pedestal side wall is penetrated.

If the lower D/W is filled with water prior to the RPV failure, particulate corium (particulate debris bed) is generated due to jet break-up at the surface of the corium jet. The corium that retains its jet form and reaches the floor of the lower D/W spreads across the floor and forms continuous corium (debris bed). The surface of the particulate debris bed and the top surface of the continuous corium can be cooled by the pool water in the lower D/W. The heat removal rate of decay heat of the continuous corium is dependent on the spreading area and composition of the corium.

On the other hand, if there is no water in the lower D/W at the time of the RPV failure, the falling corium jet does not fragment into the corium particle due to jet break-up, and spreads across the lower D/W floor. The spreading corium forms a continuous corium pool (debris bed). The heating of the atmosphere of the lower D/W by the corium pool causes the Lower Drywell Flooder System (LDF) to passively open, and the suppression pool water covers the top surface of the corium pool. Therefore, the surface is cooled. When the top surface of the continuous corium is water-cooled, it solidifies and begins to form crust. The crust has low thermal conductivity, so if a hardened thick crust is generated above the molten corium, the heat removal rate from the corium will be significantly decreased and MCCI may continue. However, a hardened crust cannot stabilise in the UK ABWR because the UK ABWR has large lower drywell floor. If a hardened thick crust is formed, it will break due to own weight and water head.

Discussion of Direct Debris Interaction

Given severe accident core melt progression, RPV breach would result in the release of molten debris to the lower drywell. If the molten debris discharged to the lower drywell deposits on the containment boundary wall such as the upper D/W wall and the access tunnel hatch, integrity of the containment may be lost. Else if the access tunnel wall or the vacuum breakers are damaged due to deposited molten debris, the pressure suppression function of the UK ABWR containment may be lost, which may cause early containment failure due to overpressure. These phenomena are defined as "Direct debris interaction". It should be noted that Molten Core Concrete Interaction (MCCI) is not included in the direct debris interaction because the concrete floor of the lower D/W is not the containment boundary and pressure boundary between the D/W and wetwell. Effectiveness of the safety system equipped in the UK ABWR is demonstrated by effectiveness evaluations, and direct debris interaction is considered as one of the criteria of effectiveness evaluations in severe accident. The item confirmed is that PCV boundary does not fail due to a direct contact of corium (containment shell attack).

In case of low pressure melt ejection scenario, the molten debris would contact only the Lower Drywell Floor (LDF) and the pedestal wall and then it would be cooled by water supplied from LDF system or lower drywell injection system. However, in case of high pressure melt ejection scenario, the molten debris might be entrained by the high pressure steam flow from the RPV breach and it might contact with the liner, lower drywell tunnels, and penetrations that could result in containment failure.

Table 25.5.2-9 Containment Failure Modes Explicitly Modelled in the CETs

Containment failure mode	Fission product Release Location	Failure Size	End State ID	Remarks
Overpressure and Overtemperature (Late)	Drywell head flange	0.068 m^2	OP/OT	-
			OP/OT_PS	With success of drywell spray
Overpressure and Overtemperature (Early)	Drywell head flange	0.068 m^2	C	With failure of reactivity control
			RR	RPV rupture
			RR_LD	RPV rupture with success of LDF
In-vessel FCI	Drywell head flange	0.68 m^2	RE	-
Ex-vessel FCI	Drywell head flange	0.68 m^2	PE	-
DCH	Drywell head flange	0.68 m^2	DCH	-
MCCI	Drywell head flange	0.68 m^2	CCI	-
Containment bypass	RHR suction line	0.0769 m^2	BYPASS	Including ISLOCA, BOC and Hydrogen combustion
Containment isolation failure	AC line	0.229 m^2	PCI	-
S/P Bypass	Drywell head flange	0.068 m^2	SPBYP	With failure of VSS
Direct Debris Interaction	Hatches in Access Tunnel	0.0383 m^2	DI	With failure of W/W to D/W vacuum breakers

Table 25.5.2-10 Intact PCV Sequence Modelled in the CETs

Intact PCV sequence	End State ID	Fission Product Release Location	Leakage Rate / Venting Line Size
Intact PCV without RPV failure	KV	D/W	0.4 percent /day (below 1Pd) 1.3 percent /day (higher than 1Pd)
Intact PCV with RPV failure	KP	D/W	
Success of filtered PCV venting without RPV failure	FVV	S/P air space	0.038 m^2
Success of filtered PCV venting with RPV failure	FVP	S/P air space	0.038 m^2
Success of no filtered PCV venting without RPV failure	VV	S/P air space	0.038 m^2
Success of no filtered PCV venting with RPV failure	VP	S/P air space	0.038 m^2

Table 25.5.2-11 Key SA Phenomena and Evaluation Method

Key Phenomena	Evaluation Method
Overpressure and Overtemperature	MAAP code is used to evaluate the containment pressure and containment temperature during a severe accident. Plant damage states are categorised into some representative sequences such as TQUV, TQUX, TB, TW, TC and AE. The plant challenges conditions due to overpressure and overtemperature are investigated based on the above representative sequences.
Ex-Vessel Fuel Coolant Interaction (FCI)	In this mode, the containment is over-pressurised by a large amount of steam or is loaded by the impulse of FCI. JASMIN and AUTODYN codes are used to evaluate the impulse load due to FCI. Separate effects calculations using MAAP input to establish initial conditions that are used to determine the dynamic pressurisation effects. The branching probability due to FCI is evaluated based on the ROAAM method [Ref-25.49]. This result is utilised to be compared with the estimated branch probability using information in NUREG/CR-4700 [Ref-25.50].
Direct Containment Heating (DCH)	The MAAP code is used to evaluate the containment pressure and containment temperature after DCH occurs. The branching probability due to DCH is evaluated based on the ROAAM method [Ref-25.49]. This result is utilised to be compared with the estimated branch probability using information in NUREG/CR-4700 [Ref-25.50].
Molten Core Concrete Interaction (MCCI)	The MAAP code is used to calculate erosion depth of basemat and pedestal wall in the lower drywell. The branching probability due to MCCI is evaluated based on the ROAAM method [Ref-25.49]. This result is utilised to be compared with the estimated branch probability using information in NUREG/CR-4700 [Ref-25.50].
Direct Debris Interaction	Direct debris interaction is considered in RPV failure scenario with high pressure and low pressure. This assessment includes the direct debris interaction with the liner, lower drywell tunnels, and penetrations that could result in containment failure.
Sub-atmospheric Containment Pressure	Two kinds of accident scenario were considered for sub-atmospheric containment pressure assessment. - Pipe rupture outside the containment - PCV pressure decreasing due to PCV cooling The SHEX and MAAP codes are used to evaluate the containment pressure during an accident.
Hydrodynamic Load	All accident scenarios including both design basis accidents and beyond design basis accidents are investigated to identify appropriate condition for hydrodynamic load assessment. They include the following consideration. (a) SRV in late phase (b) SRV in early phase (c) Containment venting (d) Piping break in late phase (e) Vessel breach

(2) Subtask 2: Containment Structural Analysis

Direct Debris Interaction

The High Pressure Melt Ejection (HPME) scenario was considered for direct debris interaction. Four kinds of structures (the upper D/W wall, access tunnel wall, access tunnel hatch and vacuum breakers) have been examined in high pressure melt ejection scenario. In case of low pressure melt ejection scenario, the molten debris would contact with only the lower drywell floor and the pedestal wall and then it would be cooled by water supplied from LDF system or lower drywell injection system.

The following conclusions were made based on the results of the containment analysis.

- The maximum molten debris height in the lower drywell was evaluated to be 0.45 m in low pressure melt ejection scenario. As the molten debris contacts only the lower drywell floor and the pedestal wall and then would be cooled by water supplied from LDF system or lower drywell injection system, it would not have any impact on PCV boundary.
- CFD analysis results indicated that molten debris does not deposit on the access tunnel hatch and V/Bs. Therefore, time dependent temperature evaluation for the access tunnel hatch and V/Bs were not conducted.
- Deposition of the molten debris on the upper D/W wall did not fail the containment wall until 35 hours after RPV failure which occurred 5 hours after the beginning of the sequence. Therefore operators would have time margins of at least 40 hours for containment spray injection into the containment to cool the upper D/W and prevent containment failure resulting from direct contact of the molten debris to the containment wall.
- Even if the molten debris enters the access tunnel, the access tunnel wall would not fail because the amount of molten debris entering the tunnel was estimated to be too small to heat sufficiently.

However, considering radiation heating from 1,000 K debris on the lower D/W structures to the access tunnel and V/B structures as the worst case, the access tunnel and the V/Bs are damaged by overtemperature. Therefore, the potential for occurrence of this severe scenario cannot be quantitatively ignored based on the current HPME knowledge for ABWR type lower D/W configuration.

Thus, it was concluded that there is potential for access tunnel and V/B failures due to overtemperature, and therefore the Level 2 PSA treats the direct debris interaction with an assumption that PCV failure occurs instantaneously if RPV failure occurs with high pressure.

Sub-atmospheric Containment Pressure

Two kinds of accident scenario were considered in the sub-atmospheric containment pressure assessment.

- Pipe rupture outside the containment

- PCV pressure decreasing due to PCV cooling

The technical basis for discounting the possibility of containment failure due to sub-atmospheric containment conditions is listed below.

- The allowable sub-atmospheric pressure of the RCCV liner and liner anchor is 325 kPa.
- In the scenario of rapid steam condensation in the containment at DBA conditions, the calculated maximum Reactor Building to Primary Containment differential pressure for this event was 10.3 kPa [diff.] which does not exceed the containment sub-atmospheric design value of 14 kPa [diff.].
- In the scenario of rapid steam condensation in the containment after PCV venting, PCV failure might occur if the PCV spray was accidentally operated with maximum flow rate for a significant period of time. However, as almost all fission product aerosols were already discharged to the environment by the time of containment venting and the containment pressure was low, the fission product mass fraction released to the environment did not increase significantly after PCV failure. This means PCV failure due to sub-atmospheric containment pressure after PCV venting does not affect the source term. Therefore, it was concluded that it was not necessary to model this failure mode in the Level 2 PSA.
- In the scenario of pipe rupture outside the containment, the calculated maximum R/B to PCV differential pressure for this event was 50 kPa [diff.] which does not exceed the maximum allowable differential pressure.

Hydrodynamic Load

All accident scenarios, including not only design basis accidents but also beyond design basis accidents, were investigated to identify the appropriate conditions for hydrodynamic load assessment. They include consideration of the following issues.

- SRV in late phase
- SRV in early phase
- Containment venting
- Piping break in late phase
- Vessel breach

The technical basis for discounting the possibility of containment failure due to hydrodynamic load is listed below.

- In the “Core injection by RCIC in the loss of Class 1 AC power” scenario, the S/P water level is higher than that postulated in DBA scenarios. Hydrodynamic loading due to SRV depressurisation might become more severe than that in DBA condition. S/P water level at the time when RPV is depressurised in the above scenario is approximately 8.6 m. Structural integrity assessment for hydrodynamic load due to SRV load was conducted considering this condition.
- In the TC scenario, the S/P water level is higher than that postulated in DBA scenario. S/P water level increases quickly during TC event. S/P water level is approximately 8.6 m at the time when the PCV pressure reaches 620 kPa [gauge]. This condition becomes same as the

conditions in the "Core injection by RCIC in the loss of Class 1 AC power" scenario. Structural integrity assessment for hydrodynamic load due to SRV load was conducted considering this condition.

- In the containment venting scenario, gases in D/W start to flow into S/C after PCV venting. Steam which flows into S/P is condensed and then steam condensation causes hydrodynamic load in the S/P. In the TQUV scenario, the maximum S/P water level is approximately 14.6 m. Structural integrity assessment for hydrodynamic load due to chugging (CH) load was conducted considering this condition.
- If SRV discharge line breaks in the gas region of the upper drywell, steam from RPV is released to D/W intermittently and CH load may occur. However, as S/P water level becomes lower than that in the containment venting scenario, hydrodynamic loading due to SRV discharge line break is enveloped by that in the containment venting scenario.
- The hydrodynamic load which may occur due to vessel breach in TQUX scenario is enveloped by the hydrodynamic load which occurs under DBA LOCA conditions. Therefore, hydrodynamic loading due to vessel breach does not need to be considered in severe accident condition.
- A structural integrity assessment was conducted for lower drywell tunnel and inside flange plate. The results of the structural integrity assessment indicated that the influence of hydrodynamic loads is small compared with other loads. In addition, it was confirmed that the stress generated at the access tunnel and penetrations were within the limits of allowable stress and the containment boundary would be maintained even if hydrodynamic load was considered in severe accident scenarios.

(3) Subtask 3: Analysis of Containment Bypass Event and Containment Isolation Failure

The following events are treated as the containment failure as containment bypass events and isolation failure.

- Interface system LOCA / Break outside of containment (Containment Bypass, i.e. PCV Bypass)

  In this mode, containment is physically intact. Fission products bypass the containment and are released to reactor building at the point where the interface system LOCA or break outside of containment occurs.

- Loss of Containment Isolation

  This mode is the loss of containment isolation at the time of core damage. The Fission Product release is from vessel or containment to reactor building when the boundary is broken, such as isolation failure of Atmospheric Control System (AC) system line.

25.5.2.4 Quantification Method

The PSA quantification for the Internal Events at Power Level 2 was carried out using the CAFTA (Computer Aided Fault Tree Analysis System) software, which has been developed by the Electric Power Research Institute (EPRI). CAFTA is a computer software program used for developing reliability models of large complex systems, using fault tree and event tree methodology.

In summary, the methodology for model integration and quantification are shown in Table 25.5.2-12.

Table 25.5.2-12 Quantification Method for Level 2 PSA [Ref-25.10]

Step	Quantification method
1	Group Level 1 PSA Accident Classes into Plant Damage States (PDSs). In this Level 2 PSA, grouping is not conducted, and core damage classes are directly used as PDSs.
2	Allocate the L1 sequences to PDSs.
3	Create a top logic fault tree that includes four types of sequence markers; Level 1 sequence No., Level 2 sequence No., PDS and Release category.
4	Merge the system model fault tree files into the top logic master fault tree. a. Check for any circular logic and break it if identified. b. Apply all the CCF models built in individual fault trees.
5	Check for any model inconsistencies and fix the identified issues, including any issues with fault tree logic and reliability data.
6	Set configuration options using the master flag files, and recovery file generated for the master model.
7	Perform quantification of a top logic fault tree by FTREX.

25.5.3 Source Term Analysis

25.5.3.1 Purpose of Source Term Analysis

Source term analysis has been performed to determine the magnitude and characteristics of the source term used in the Level 3 PSA.

25.5.3.2 Evaluation Method for Source Term Analysis

The MAAP code is used for source term analysis of the UK ABWR [Ref-25.49]. The MAAP code models the significant accident phenomena that might occur in the RPV, in the containment, and in the reactor building. The MAAP code calculates the progression of postulated accident sequences from a set of initiating events to either a safe, stable state or a degraded containment condition due to overpressure or overtemperature, and fission product releases to the environment.

25.5.3.3 Release Category and Representative Accident Sequence

Nineteen end state IDs are defined in Table 25.5.2-9 and Table 25.5.2-10, and these have been modelled in the CETs. The containment response to a severe accident is depicted by the end states. These end states become the “release categories” that are used to characterise potential source terms. This section summarises the definition of the release categories to consider the magnitude and the timing of fission product releases based on the characteristics of PDSs and the end states in CETs.

(1) Grouping of PDSs

Nine PDS groups have been defined based on the following characteristics:

- PDSs with specific release characteristics (e.g. Containment bypass, S/P bypass),
- PDSs with containment failure prior to core damage (e.g. TW, AC and so on),
- Timing of fission product release, and
- S/P scrubbing effect (S/P scrubbing effect is credited when SRVs are open in non-LOCA.).

Table 25.5.3-1 includes the detailed descriptions of the nine PDS groups.

(2) Release Category and Representative PDS

Twenty-three release categories are defined considering the combination of PDS groups and end states in Level 2 PSA. In order to perform the source term analysis, specific representative accident sequences are determined based on the severity of source term (release timing and fission product release fraction) for each release category.

25.5.3.4 Source Term Analysis Conditions

Table 25.5.3-2 summarises the general conditions assumed in the source term analysis, whilst the analysis cases for each release category are described in Table 25.5.3-3. The fission product aerosol removal processes such as deposition, impaction and thermophoresis are considered in all compartments including the reactor building. The reactor building is modelled as one compartment, which does not consider inner rooms and inside walls.

25.5.3.5 Source Term Analysis Results

Table 25.5.3-4 provides a summary of the source term analysis results. As a result of source term analysis, each release category are assigned to the large release or early release as shown in Table 25.5.3-5. The large release and early release are assumed as follows [Ref-25.74].

- A release category with CsI release fraction greater than 10 percent is regarded as large release.
- A release category with the following FP release is regarded as early release.
  - Containment failure within 4 hours from RPV breach.
  - Containment failure before RPV breach and within 10 hours from an initiating event.

Table 25.5.3-1 Characteristics of PDS Group

PDS Group	PDS	Containment bypass or VSS failure	Containment failure prior to core damage	Timing of core damage*4	S/P scrubbing effect*1
PDS I	TQUV, TBP, TC-LP	No	No	Early /Late *5	High
PDS II	AE, TQUX, TBU, TBD, TC-HP, S12UX	No	No	Early /Late *5	Low
PDS III	TB	No	No	Late	Low
PDS IV	AC, TC, S12C	No	Yes	Early	Low
PDS V	TW, AW-LP, S12W	No	Yes	Late	Low
PDS VI	TW-LP	No	Yes	Late	High
PDS VII	S3UX, S3E, AN*2,	Yes (Containment bypass)	Yes	Early	Low
PDS VIII	TNQUV, TNQUX, TBPN, TBDN, TCN	Yes (VSS failure)	Yes *3	Early	Low
PDS IX	S4	No	Yes	Early	Low

Notes:

*1 S/P scrubbing is effectively credited when SRVs continue to open at debris in-vessel phase in non-LOCA. It is conservatively judged whether SRVs are opened at the timing of core damage to avoid complexity.

*2 AN is LOCA with failure of W/W to D/W vacuum breaker, resulting in sub-atmospheric pressure containment failure due to containment spray. AN is included in containment bypass since the failure location is uncertain.

*3 The timing of core damage and containment failure is almost the same. Therefore, PDSs with VSS failure are considered as the containment failure prior to core damage.

*4 The criterion of timing of core damage is the follows.

1) Containment failure within 4 hours from RPV breach,

2) Containment failure before RPV breach and within 10 hours from an initiating event.

*5 It depends on the accident progression after core damage.

Table 25.5.3-2 Summary of Source Term Analysis Conditions

Items	Conditions	Remarks
Initial reactor power	3,926 MWt	Rated power
Initial core flow rate	52.2×10^3 t/h	Rated flow rate
Initial reactor pressure	7.07 MPa[gauge]	Rated pressure
Initial reactor water level	Normal level	Normal water level
Free volume of PCV	D/W Open Space : 7,350 m^3 W/W Open Space : 5,960 m^3	Design value
Suppression pool water volume	3,580 m^3	Design value
Initial water temperature of S/P	35 ^oC	Maximum temperature during normal operation
Ultimate pressure of PCV	Pressure : 620 kPa[gauge]	Twice maximum design pressure of PCV
Ultimate temperature of PCV	Temperature: 200 °C	Judged by PCV gas temperature
Decay heat power	ANSI/ANS-5.1-1979	MAAP model
Fuel type	GE-14	10x10 fuel rods
Number of core nodes	Radial Nodes : 5 rings Axial Nodes : 28 nodes	MAAP standard setting

Table 25.5.3-3 Analysis Case and Analysis Condition for Source Term Analysis

No.	Release Categoty	Representative accident sequence	Sub-criticality	RPV				PCV						FP release		
				RCIC/HPCF	Depressurisation	LPFL	Alternative Water injection	Injection to lower D/W before RPV failure	Injection to lower D/W after RPV failure	Activation of lower D/W flooder	Alternative D/W spray	Restoration of RHR	PCV venting	PCV leakage	PCV venting or failure location	PCV venting or failure area
1	Containment Leakage from D/W	TQUV	O	X	O	X	X	X	X	O	O	O	-	D/W	-	-
2	Containment Venting	TQUV	O	X	O	X	X	X	X	O	X	X	O	D/W	W/W	0.038 m^2
3	Filtered Containment Venting	TQUV	O	X	O	X	X	X	X	O	X	X	O	D/W	W/W	0.038 m^2
4	Early Containment Failure	AC	X	O	-	X	X	X	X	O	X	X	X	D/W	D/W	0.068 m^2
5-1	Late Containment Failure	TQUV	O	X	O	X	X	X	X	O	X	X	X	D/W	D/W	0.068 m^2
5-2		AE	O	X	-	X	X	X	X	O	X	X	X	D/W	D/W	0.068 m^2
5-3		AW-LP	O	X	-	O	X	X	X	O	X	X	X	D/W	D/W	0.068 m^2
5-4		TW-LP	O	O	O *L1+30sec with 7 valves	O	X	X	X	O	X	X	X	D/W	D/W	0.068 m^2
6	Late Containment Failure with PCV spray	AE	O	X	-	X	X	X	X	O	O	X	X	D/W	D/W	0.068 m^2
7-1	In-vessel Fuel-Coolant Interaction	TQUV	O	X	O	X	X	X	X	O	X	X	X	D/W	D/W	0.68 m^2
7-2		AE	O	X	-	X	X	X	X	O	X	X	X	D/W	D/W	0.68 m^2
8-1	Ex-vessel Fuel-Coolant Interaction	TQUV	O	X	O	X	X	O	X	O	X	X	X	D/W	D/W	0.68 m^2
8-2		AE	O	X	-	X	X	O	X	O	X	X	X	D/W	D/W	0.68 m^2
9	Direct Containment Heating	TQUX	O	X	X	X	X	X	X	O	X	X	X	D/W	D/W	0.068 m^2
10-1	PCV Isolation Failure	TQUV	O	X	O	X	X	X	X	O	X	X	X	D/W	D/W	0.229 m^2
10-2		AE	O	X	-	X	X	X	X	O	X	X	X	D/W	D/W	0.229 m^2
11-1	Molten Core Concrete Interaction	TQUV	O	X	O	X	X	X	X	X	X	X	X	D/W	D/W	0.068 m^2 $\rightarrow$0.68 m^2
11-2		AE	O	X	-	X	X	X	X	X	X	X	X	D/W	D/W	0.068 m^2 $\rightarrow$0.68 m^2
12	RPV rupture	S4	O	X	-	X	X	X	X	O	X	X	X	D/W	D/W	0.068 m^2
13	Containment Bypass	S3E	O	X	-	X	X	X	X	O	X	X	X	D/W	D/W	0.0769 m^2
14	S/P Bypass	TNQUV	O	X	O	X	X	X	X	O	X	X	X	D/W	D/W	0.068 m^2
15	Direct Debris Interaction	TQUX	O	X	X	X	X	X	X	O	X	X	X	D/W	D/W	0.006 m^2 $\rightarrow$0.074 m^2
16	Long-term SBO (In-vessel FCI)	TB	O	O	X	X	X	X	X	O	X	X	X	D/W	D/W	0.68 m^2

O : Success, X : Failed, - : Not applicable

Table 25.5.3-4 Summary of Key Event Timing and Fission Product Mass Fraction Released to the Environment

No.	Release Categoty	Representative accident sequence	Event (hr)			FP mass fraction released to environment (-)											
			Core Damage	RPV failure	PCV failure / PCV venting	Inert	CsI	TeO2	SrO	MoO2	CsOH	BaO	La2O3	CeO2	Sb	Te2	UO2
1	Containment Leakage from D/W	TQUV	0.6	6.9	-	2.2E-02	9.3E-10	1.3E-10	2.1E-11	7.5E-13	1.7E-09	1.0E-11	1.3E-12	9.9E-12	4.4E-09	2.3E-11	8.5E-14
2	Containment Venting (2Pd) NoDWS	TQUV	0.6	6.9	12.8	1.0E+00	6.6E-04	3.7E-05	1.3E-07	3.0E-08	7.8E-04	1.1E-07	1.0E-08	6.6E-08	4.6E-04	5.4E-06	5.7E-10
2a	Containment Venting (2Pd)	TQUV	0.6	6.9	35.2	7.8E-01	4.4E-08	8.6E-09	2.1E-11	2.3E-12	4.6E-06	1.3E-11	1.4E-12	1.0E-11	8.6E-06	1.7E-06	8.6E-14
3	Filtered Containment Venting	TQUV				DF of filtered containment venting system will be considered for the source term of Case 2.											
4	Early Containment Failure	AC	1.5	9.4	1.0	1.0E+00	6.3E-01	5.1E-01	1.0E-02	1.5E-02	5.0E-01	1.3E-02	7.9E-04	5.4E-03	4.4E-01	1.2E-02	4.3E-05
5-1	Late Containment Failure	TQUV	0.6	6.9	12.8	1.0E+00	7.7E-04	2.4E-04	5.3E-07	9.5E-08	4.1E-02	3.7E-07	3.4E-08	2.4E-07	2.8E-02	3.8E-06	2.0E-09
5-2		AE	0.2	6.6	7.4	9.9E-01	1.2E-01	4.2E-02	3.8E-05	1.7E-05	1.4E-01	1.9E-05	1.7E-06	1.2E-05	3.4E-01	1.9E-05	1.0E-07
5-3		AW-LP	18.8	32.5	15.2	1.0E+00	6.7E-01	5.6E-01	7.2E-03	2.2E-02	5.5E-01	1.8E-02	3.6E-04	2.1E-03	4.8E-01	1.2E-03	1.5E-05
5-4		TW-LP	21.9	35.5	16.6	1.0E+00	2.4E-02	1.7E-02	2.4E-03	1.8E-04	4.7E-02	1.1E-03	1.9E-04	1.3E-03	1.3E-01	8.5E-04	1.2E-05
5a		TQUX	0.6	4.6	15.1	1.0E+00	1.4E-01	3.9E-02	5.4E-09	2.0E-08	4.1E-02	3.6E-08	5.4E-10	4.1E-09	1.0E-04	0.0E+00	0.0E+00
6	Late Containment Failure (D/W failure with PCV spray success)	AE	0.2	6.6	37.5	1.0E+00	5.7E-04	1.8E-04	6.2E-08	3.8E-07	2.6E-02	3.0E-07	4.0E-09	1.1E-08	6.2E-05	8.0E-06	6.9E-11
6a		TQUV	0.6	6.9	35.2	1.0E+00	2.2E-06	1.7E-06	1.5E-08	5.2E-10	7.1E-03	7.1E-09	9.5E-10	7.0E-09	1.3E-03	1.9E-06	6.0E-11
7-1	In-vessel Fuel-Coolant Interaction (FCI)	TQUV	0.5	6.7	3.1	1.0E+00	6.3E-02	3.6E-02	1.4E-03	9.1E-03	1.9E-01	7.5E-03	1.4E-04	4.6E-04	1.2E-01	6.0E-06	3.6E-06
7-2		AE	0.2	6.4	2.1	8.5E-01	3.5E-01	3.5E-01	1.5E-03	6.0E-03	3.6E-01	7.6E-03	1.8E-04	5.0E-04	4.8E-01	3.0E-04	4.3E-06
8-1	Ex-vessel Fuel-Coolant Interaction (FCI)	TQUV	0.6	6.9	6.9	1.0E+00	4.5E-03	1.9E-03	1.9E-06	1.3E-05	1.2E-02	1.8E-05	3.2E-07	3.6E-07	3.0E-02	0.0E+00	0.0E+00
8-2		AE	0.2	6.6	6.6	8.4E-01	1.5E-01	7.7E-02	1.5E-06	3.3E-05	1.6E-01	1.1E-05	1.9E-07	3.7E-07	3.7E-01	0.0E+00	0.0E+00
9	Direct Containment Heating (DCH)	TQUX	0.6	4.6	4.6	1.0E+00	2.1E-01	6.9E-02	8.0E-07	1.4E-06	4.7E-02	5.3E-06	6.1E-08	6.0E-07	5.4E-04	0.0E+00	0.0E+00
10-1	PCV Isolation Failure	TQUV	0.6	6.8	-	1.0E+00	2.3E-03	4.1E-03	5.6E-05	1.0E-05	2.1E-02	3.4E-05	4.7E-06	3.2E-05	2.7E-02	2.3E-05	2.9E-07
10-2		AE	0.2	6.8	-	9.7E-01	3.7E-01	3.0E-01	2.8E-03	2.8E-02	3.0E-01	1.2E-02	2.0E-04	6.3E-04	6.3E-01	1.1E-04	4.1E-06
11-1	Molten Core Concrete Interaction (MCCI)	TQUV	0.6	6.9	16.7	1.0E+00	1.4E-02	5.0E-03	6.6E-07	2.3E-09	3.4E-02	5.6E-06	3.6E-08	3.1E-07	9.5E-02	2.7E-03	5.9E-08
11-2		AE	0.2	6.6	7.3	8.7E-01	1.9E-01	2.7E-02	6.0E-03	7.7E-06	1.4E-01	2.6E-03	2.4E-04	2.4E-03	4.6E-01	4.8E-03	1.5E-05
12	RPV rupture	S4	0.0	0.0	0.0	9.8E-01	3.8E-01	2.4E-01	1.3E-03	5.8E-03	1.9E-01	6.2E-03	6.0E-05	8.4E-05	5.0E-01	0.0E+00	0.0E+00
13	Containment Bypass (RHR)	S3E	0.2	6.7	-	1.0E+00	9.0E-01	7.7E-01	7.1E-03	6.4E-02	8.2E-01	3.3E-02	3.7E-04	7.5E-04	6.4E-01	2.8E-06	7.7E-08
14	S/P Bypass	TNQUV	0.6	7.1	0.6	1.0E+00	1.8E-01	1.9E-01	1.2E-03	6.6E-03	1.7E-01	6.2E-03	8.0E-05	1.8E-04	1.3E-01	0.0E+00	0.0E+00
15	Direct Debris Interaction	TQUX	0.6	4.6	4.6	1.0E+00	2.8E-02	6.6E-02	6.9E-07	1.2E-06	5.6E-02	4.6E-06	5.2E-08	5.2E-07	5.7E-04	0.0E+00	0.0E+00
16	Long-term SBO(In-vessel FCI)	TB	11.8	20.8	16.6	1.0E+00	4.9E-01	5.1E-01	4.5E-04	3.9E-03	3.6E-01	4.5E-03	9.3E-05	9.7E-05	1.0E-01	0.0E+00	0.0E+00
16a	(Late containment failrue)	TB	11.8	18.5	18.5	1.0E+00	2.2E-01	2.2E-01	3.3E-06	6.8E-06	1.8E-01	2.0E-05	3.9E-07	3.5E-06	9.5E-04	0.0E+00	0.0E+00
16b	(PCV isolation failure)	TB	11.8	18.2	-	1.0E+00	3.0E-01	1.2E-01	4.2E-06	9.4E-06	1.5E-01	2.7E-05	4.2E-07	3.8E-06	1.3E-03	0.0E+00	0.0E+00

Table 25.5.3-5 Summary of Release Category (1/2)

No.	Release Category	Representative PDS	Designator	Release Fraction of CsI	Large	Early
1	Containment Leakage (KV and KP_I, II and III)	TQUV	KP	9.3E-10	N	-
2	Containment Venting (VV and VP_I, II and III)	TQUV	VP	4.4E-08	N	-
3	Filtered Containment Venting (FVV and FVP_I, II and III)	TQUV	FVP	4.4E-08 *	N	-
4	Early Containment Failure (ALL_IV_except_BYPASS)	AC	C	6.3E-01	Y	Y
5-1	Late Containment Failure (OP/OT_I)	TQUV	OP/OT1	7.7E-04	N	-
5-2	Late Containment Failure (OP/OT_II)	AE	OP/OT2	1.2E-01	Y	Y
5-3	Late Containment Failure (ALL_V_except_BYPASS)	AW-LP	OP/OT3	6.7E-01	Y	N
5-4	Late Containment Failure (ALL_VI_except_BYPASS)	TW-LP	OP/OT4	2.4E-02	N	-
6	Late Containment Failure (OP/OT_PS_I, II and III)	AE	OP/OT_PS	5.9E-04	N	-
7-1	In-vessel Fuel-Coolant Interaction (RE_I)	TQUV	RE1	6.3E-02	N	-
7-2	In-vessel Fuel-Coolant Interaction (RE_II)	AE	RE2	3.5E-01	Y	Y
8-1	Ex-vessel Fuel-Coolant Interaction (PE_I)	TQUV	PE1	4.5E-03	N	-
8-2	Ex-vessel Fuel-Coolant Interaction (PE_II)	AE	PE2	1.5E-01	Y	Y
9	Direct Containment Heating (DCH_II)	TQUX	DCH	2.1E-01	Y	Y
10-1	PCV Isolation Failure (PCI_I)	TQUV	PCI1	2.3E-03	N	-
10-2	PCV Isolation Failure (PCI_II)	AE	PCI2	3.7E-01	Y	Y

Table 25.5.3-5 Summary of Release Category (2/2)

No.	Release Category	Representative PDS	Designator	Release Fraction of CsI	Large	Early
11-1	Molten Core Concrete Interaction (CCI_I)	TQUV	CCI1	1.4E-02	N	-
11-2	Molten Core Concrete Interaction (CCI_II)	AE	CCI2	1.9E-01	Y	Y
12	RPV Rupture (ALL_IX_except BYPASS and RR and RR_LD_AE)	S4	RR	3.8E-01	Y	Y
13	Containment Bypass (ALL VII and BYPASS_except III)	S3E	BYPASS	9.0E-01	Y	Y
14	S/P Bypass (ALL_VIII_except_BYPASS)	TNQUV	SPBYP	1.8E-01	Y	Y
15	Direct Debris Interaction (DDI_II)	TQUX	DDI	8.8E-03	N	N
16	Long term SBO (Containment Failure w/o Spray _III)*	TB (In-vessel FCI)	LTSBO	4.9E-01	Y	N

* DF of filtered containment venting system is considered in the Level 3 PSA.

25.6 Level 3 PSA for Internal Events at Power

The methodology adopted for the IEAP events is described in this section as an illustration of the general principles. The frequencies of the release categories, consisting of accident sequences involving fuel melt with similar radiological consequences are identified at the Level 2 PSA stage. The Level 2 PSA for IEAP is described in Section 25.5.

Other initiating event groups may also lead to environmental releases and offsite consequences; the Level 3 methodologies and results are presented and discussed in subsequent sections as follows:

- IE at Shutdown (Section 25.8),
- Spent Fuel Storage Pool (SFP) IE (Section 25.9),
- Events induced by Internal Hazards (Section 25.10),
- Events induced by External Hazards including seismic (Section 25.11),
- Fuel route events (Section 25.12), and
- Non-reactor faults (Section 25.13).

25.6.1 Assessment Targets for Level 3 PSA

The ONR SAPs set out the following Basic Safety Level (BSL) and Basic Safety Objective (BSO) levels of individual risk (Target 7), facility dose bands (Target 8) and societal risk (Target 9):

- **Individual risk (Target 7) -** The targets for the individual risk of death to a person at an offsite location, from an accident that results in exposure to ionising radiation, are:
  - BSL: 1 x 10^{-4} pa
  - BSO: 1 x 10^{-6} pa

- **Facility dose bands (Target 8) -** The targets for the total predicted frequencies of accidents on an individual facility, which could give doses to a person off the site are:

Effective dose [mSv]	Total predicted frequency per annum	
	BSL	BSO
0.1 to 1	1	$1x10^{-2}$
1 to 10	$1x10^{-1}$	$1x10^{-3}$
10 to 100	$1x10^{-2}$	$1x10^{-4}$
100 to 1,000	$1x10^{-3}$	$1x10^{-5}$
> 1,000	$1x10^{-4}$	$1x10^{-6}$

- **Societal risk (Target 9) -** The targets for the total risk of 100 or more fatalities, either immediate or eventual, from accidents at the site resulting in exposure to ionising radiation, are:
  - BSL: 1 x 10^{-5} pa
  - BSO: 1 x 10^{-7} pa

These Targets are similarly adopted in BP14.3 of the Hitachi-GE Nuclear Safety and Environmental Design Principles (NSDEPs) for UK ABWR [Ref-25.58].

The Level 3 PSA assessments performed focus on offsite radiological consequences (although Target 9 also needs to consider the potential for fatalities on-site).

For assessment against facility dose bands (Target 8) deterministic dose calculations were performed, which consider the consequences from a single set of atmospheric conditions. Whereas, for assessment against risk targets (Target 7 and 9), probabilistic risk calculations were performed to consider the consequences over a wide range of atmospheric conditions sampled from a meteorological database.

Figure 25.6.1-1 below shows a schematic representation of the process from the Level 1 PSA outputs to the assessments against Targets 7, 8 and 9. It is noted that the route to process Level 1 PSA 'success' sequence groups is not shown in Figure 25.6.1-1 but this follows the same process.

The following subsections provide summaries of the key inputs, assumptions and methodologies employed in the assessment.

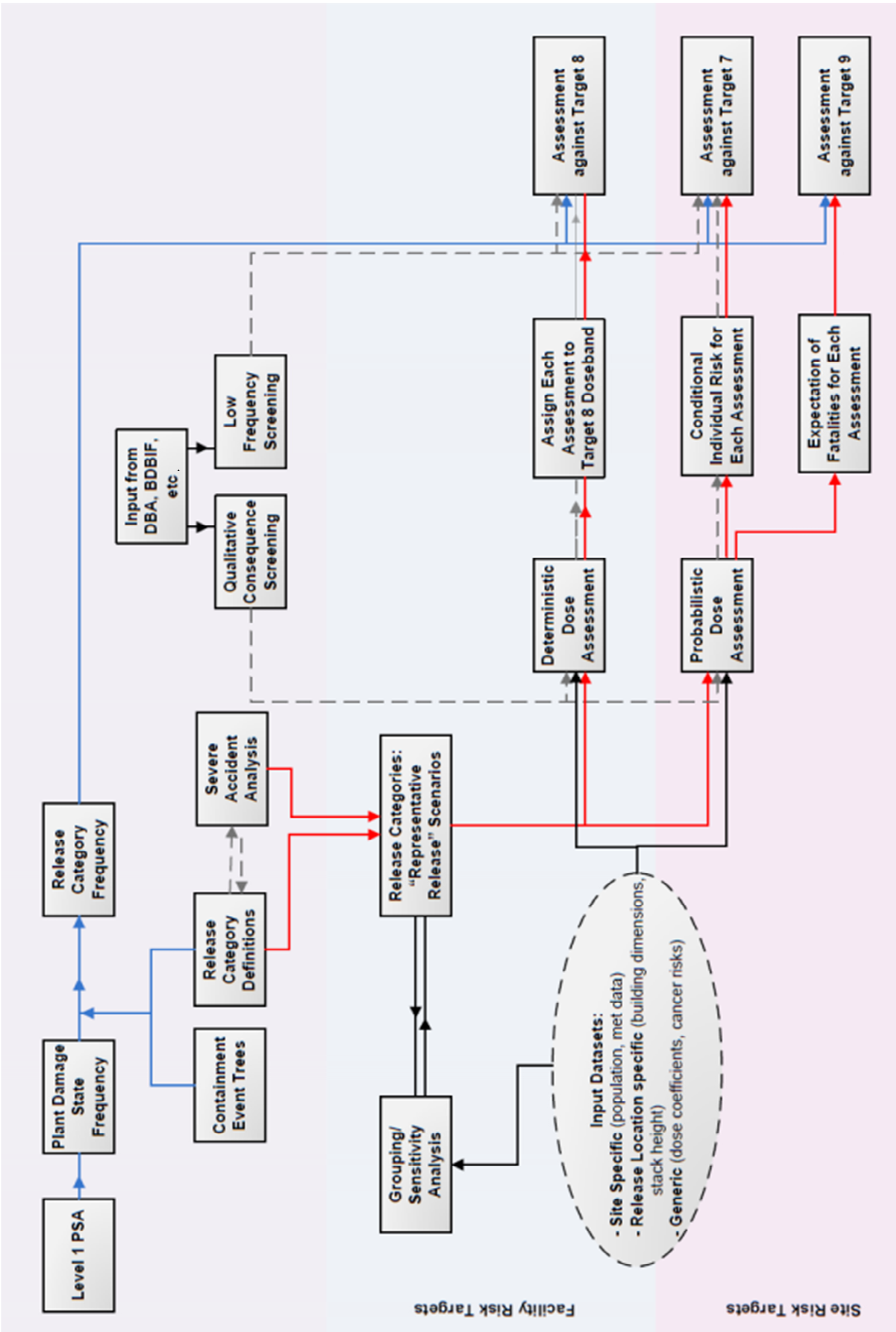

Figure 25.6.1-1 Combined Roadmap for Assessment of Postulated Severe Accidents against Numerical Targets

25.6.2 Interface between Level 2 and Level 3 PSA

Release Categories (RCs) and their calculated conditional frequencies were determined for IEAP PSAs during the Level 2 PSA stage. The magnitude and characteristics of the fission product mass releases associated with each RC are predicted for representative severe accident sequences using the MAAP code as part of the SAA [Chapter 26 of PCSR Rev. C].

The RCs and representative SAA sequences leading to a degraded fuel are summarised below.

25.6.2.1 Radionuclide release categories

The Level 2 PSA CETs describe the response of the containment to a severe accident, based on the consideration of a number of phenomenological events, containment integrity status, release timing, failure/release location, mitigation effects and etc. These CETs are grouped into the RCs that are used to characterise potential source terms.

Sixteen initial RCs were defined for the interface between Level 2 PSA and Level 3 PSA for IE at Power. The RCs, along with their frequency and the PDS condition assumed for the representative SAA sequences, are listed in Table 25.6.2-1.

Multiple representative SAA sequences were adopted for RCs 5, 7, 8, 10 and 11 to capture the differences in accident progression behaviours, i.e. timing of containment and core failures, respectively. As a result, a total of 23 release categories were evaluated in the Level 3 PSA for IEAP. These are also referred to as Level 3 PSA cases and, for clarity in presenting results graphically in section 25.15, the release category identifier is preceded by the letter “P”.

Table 25.6.2-1 Release Categories, Frequency of Occurrence and PDS for Representative SAA Sequence for IE at Power

Level 3 PSA case / RC	Release Category description (PDS of representative SAA sequence)	Frequency (/y)
P1	Containment Leakage from D/W - failed RPV (TQUV)	3.84E-08
P2	Containment Venting (TQUV no DW sprays)	1.05E-09
P3	Filtered Containment Venting (TQUV no DW sprays)	1.31E-07
P4	Early Containment Failure (AC)	1.54E-08
P5-1	Late Containment Failure (TQUV)	4.39E-09
P5-2	Late Containment Failure (AE)	6.33E-11
P5-3	Late Containment Failure (AW-LP)	3.47E-10
P5-4	Late Containment Failure (TW-LP)	3.93E-09
P6	Late Containment Failure with PCV spray (AE)	4.21E-09
P7-1	In-vessel Fuel-Coolant Interaction (TQUV)	5.41E-12
P7-2	In-vessel Fuel-Coolant Interaction (AE)	4.22E-13
P8-1	Ex-vessel Fuel-Coolant Interaction (TQUV)	2.51E-10
P8-2	Ex-vessel Fuel-Coolant Interaction (AE)	1.50E-11
P9	Direct Containment Heating (TQUX)	2.41E-11
P10-1	PCV Isolation Failure (TQUV)	2.28E-10
P10-2	PCV Isolation Failure (AE)	8.06E-11
P11-1	Molten Core Concrete Interaction (TQUV)	1.91E-09
P11-2	Molten Core Concrete Interaction (AE)	1.77E-11
P12	RPV rupture (S4)	1.00E-08
P13	Containment Bypass (S3E)	1.85E-08
P14	S/P Bypass (TNQUV)	1.75E-10
P15	Direct Debris Interaction (TQUX)	2.70E-09
P16	Long Term SBO (TB in-vessel FCI)	1.58E-09
Total Frequency for IE at Power release categories		**2.34E-07**

(1) Mass fraction releases and release profiles

The SAA code MAAP 4.07 was used to predict the magnitudes and time-dependent releases of fission products from the core and to model radionuclide transport mechanisms in plant systems, leading to predictions of releases to the environment. The SAA predicts releases to the atmosphere via the main stack or blowout panels as gas / steam mixtures. Water accumulated in pools and sumps is modelled as retained within the building structures, so no releases of contaminated water are predicted.

Core mass fraction released to the environment was generated for 12 fission product groups as function of time over a period of 72 hours for each of the RCs for IEAP.

Although MAAP produces data for a 72 hour, the actual duration of the release may be less than this since the release may not start for some hours after the start of the accident transient and there may be no significant further release after a certain time. Nevertheless, significant releases may occur over periods of tens of hours during which the meteorological conditions – in particular the wind direction – may change. The offsite accident consequence assessment code used in the risk analysis – PC-COSYMA – can model the release as up to six phases, each of which can have different meteorological conditions.

For each fission product group in MAAP, the time dependent release data are converted into a matrix of mass fractions released in a number of distinct release phases, typically between 4 and 6 phases. The mass fractions released in each phase are conservatively assumed to occur over one hour at the start of the phase, regardless of the duration of the phase. Equal duration phases could be used but, in order to try and reproduce the actual release profile more accurately, the start of the release phases are typically defined by 'events', such as:

- Start of SAA transient (representing time relative to reactor trip),
- Onset of core melt,
- Release amount of the 'main group' reaches 10 percent of its final release amount,
- Release amount of the 'main group' reaches 50 percent of its final release amount,
- 12 hours following onset of core melt,
- Remainder of Release, i.e. residual release following the last of the above events, and
- Late Release, i.e. an optional phase for SAA sequences with a distinct late release.

The 'main group' is selected as the fission product group for which the 'events' give a sensible division into phases. For most RCs and in most cases, this is the noble gas group. Where two or more 'events' happen within the same hour, they are condensed into a single phase.

Phases that are wholly released via the Standby Gas Treatment System (SGTS), (unfiltered) hardened containment vent line or Filtered Containment Venting System (FCVS) are identified and modelled as stack releases in the dispersion calculations. Other phases are conservatively modelled as releases into the building wake, even if a portion of the release may be via the stack. Filter Decontamination Factors (DFs) are applied for releases via the FCVS or SGTS. The DFs for the SGTS are modelled in MAAP 4.07 whilst those for the FCVS are applied by post-processing the release fraction values.

(2) Releases of radioactivity to the environment

The 12 fission product groups from MAAP 4.07 are condensed into 10 element-based Source Term Groups (STGs) which are shown in Table 25.6.2-2. For each RC, the release of radioactivity to the environment is determined from:

- The matrix of mass release fractions for each of the STGs in each release phase,
- The list of elements assigned to each of the 10 STGs, and
- The reference whole core inventory.

This data is used in two ways:

(i) For the deterministic dose calculations, the mass fraction matrix is used to derive the total releases (in Bq) for each radionuclide, based on the element list and taking into account radioactive decay and in-growth in the period prior to release of each phase. These are calculated in a pre-processor step which uses the reference core inventory.

(ii) The matrix of mass release fractions and element list is used directly in the input for the probabilistic calculations. The reference core inventory is separately input to PC COSYMA which performs the decay and in-growth calculations.

Two sets of releases to the environment have been defined for IEAP, based on assumptions regarding iodine chemistry which are intended to ‘bracket’ the likely conditions. These are called the Base Case and the Modified Case, and these have been used in a number of sensitivity calculations.

- The Base Case is a high pH iodine approach which considers the iodine releases predicted in the SAA to be a reasonable representation of iodine behaviour. This is the case where the water, particularly in the Suppression Pool, is maintained at a pH ≥ 7. In this pH condition, production of elemental iodine is suppressed and the release of MAAP Group 2 is used to directly represent the release of iodine. Consequently, the Base Case source term contains iodine in its aerosol form, which would be the predominant form in high pH conditions. A predetermined nominal DF value is applied for releases via SGTS and FCVS.
- The Modified Case is an empirically based iodine chemistry approach in which the NUREG-1465 [Ref-25.59] methodology for iodine chemistry is implemented in conservative manner. This approach is appropriate where the pH of water pools is not strictly controlled at a pH ≥ 7. It accounts for oxidation of iodine in the water pool and reactions to form more volatile iodine species in the longer term. For the Modified Case, iodine release fractions are calculated as follows:
  - The mass release fraction of MAAP Group 2 (CsI) is used as the release fraction of iodine in aerosol form for each phase.
  - The iodine mass release fraction in each phase is then increased to allow for formation of iodine in vapour phase species (assumed to be elemental iodine and methyl iodide) prior to release. These vapour species are conservatively assumed to be present immediately after the fuel starts to melt and then to behave like noble gases and not deposit out in the containment before release. The additional vapour species are, therefore, expressed as a fraction of MAAP Group 1 (noble gases):
    (i) 0.0485 multiplied to release fraction of noble gas is added to the source term as elemental iodine (I_2),
    (ii) 0.0015 multiplied to release fraction of noble gas is added to the source term as methyl iodide (CH_3I).
  - The results presented in this report are for the Modified Case unless specifically stated as results for the Base Case.

Table 25.6.2-2 Representation of Fission Product Elements in Terms of PC COSYMA Source Term Groups

Fission product elements	Source Term Group
Krypton, Xenon	Group 1
Iodine	Group 2
Tellurium	Group 3
Strontium	Group 4
Molybdenum, Ruthenium, Technetium, Rhodium	Group 5
Caesium, Rubidium	Group 6
Barium	Group 7
Lanthanum, Americium, Curium, Yttrium, Zirconium, Niobium, Neodymium, Praseodymium	Group 8
Cerium, Neptunium, Plutonium	Group 9
Antimony	Group 10

25.6.3 Deterministic Dose Calculations

Deterministic dose calculations, for assessment against facility dose bands (Target 8), were performed using a spreadsheet codification of the UK Gaussian plume model, subsequently referred to as PUMA. The spreadsheet code also incorporates:

- Dose coefficients for internal exposure from intakes of radioactivity taken from publications by the International Commission on Radiological Protection (ICRP) [Ref-25.60],
- Dose coefficients for external exposure based on the Health Protection Agency (HPA now subsumed into Public Health England, PHE) 'GRANIS model' [Ref-25.61] and US EPA Federal Guidance Report No. 12 (FGR12) [Ref-25.62], and
- Generalised UK habits data taken from NRPB-W41 [Ref-25.63].

The key model parameters, assumptions and inputs are summarised below.

25.6.3.1 Atmospheric Dispersion Methodology

The dispersion of radionuclides released to the atmosphere in the Level 3 PSA cases 1 was modelled using a 'modified R-91' approach, which is an implementation of the National Radiological Protection Board's (NRPB) R91 Gaussian plume model [Ref-25.64] and its supplementary models – the NRPB-R157 (for building wake effects) [Ref-25.65] and NRPB-R122 (for dry and wet deposition from the plume) [Ref-25.66]. These dispersion-related models are considered to be valid for distances ranging from about 100 m to about 100 km. Radionuclides are assumed to be released from either the stack (57 m) or into the building wake.

The HPA has published the results of an intercomparison study which compared the 'modified R-91' Gaussian plume model with the more sophisticated UK Meteorological Office's Lagrangian model NAME III [Ref-25.67]. The study concluded that the 'modified R-91' model generally gives conservative results for the plume centreline Time Integrated Air Concentration (TIAC) in the near field (e.g. about a factor of 3 at 1 km for Pasquill stability category D).

A release duration factor was used to model the effect of plume spread in the crosswind direction due to wind meander. i.e. an increase in the σ_y term values for release durations greater than 1 hour, as described in [Ref-25.64]. For deterministic assessments, a single phase release over a period of 4 hours was assumed for all RCs. This is considered to reasonably represent wind meander factors.

Pasquill stability category D and a windspeed of 5 m/s (the 'D5' weather condition) were adopted for the deterministic calculations. The 'D5' weather condition is consistent with typical or best-estimate weather conditions for the site.

Rainfall was assumed to not occur for the duration of the plume passage. Site meteorological data indicates that this assumption (i.e. 0 mm/hr rainfall rate) persists for approximately 68 percent of the time, so again this is consistent with typical or best-estimate weather conditions.

Four general physicochemical forms of material were considered in the deterministic dispersion calculations:

- Noble gases that do not deposit (MAAP Group 1 and STG 1),
- Aerosols that are assumed to behave in the same way regardless of chemical composition (MAAP Groups 3 to 12 and STGs 3 to 10),
- Iodine (MAAP fission product Group 2 and STG 2) released in the form of CsI and assumed to behave as an aerosol (as above),
- Elemental Iodine (I_2) that has distinct dry and wet deposition velocities; the dry deposition velocity is higher than that for aerosols reflecting its reactive nature, and

- Organic iodide (assumed to be methyl iodide); the dry deposition velocity is lower than that for aerosols reflecting its unreactive nature.

25.6.3.2 Exposure Pathways and Habit Assumptions

Effective doses to representative members of the public were calculated as the sum of:

- The external gamma irradiation by radioactive material entrained in the passing plume (often referred to as cloud gamma or cloud shine),
- The internal irradiation due to inhalation of radioactive material entrained in the passing plume, and
- The long term (50 years) external gamma irradiation from radionuclides deposited on the ground (often referred to as ground gamma or ground shine).

The effective doses were typically calculated for an adult but, where the dose from these three pathways was < 1Sv, additional sensitivity calculations were performed for 10 years old children and infants.

The individual effective dose was calculated at distances of 400 m, 700 m, 1,000 m and 1,500 m downwind of the release point to illustrate the variation of dose with distance over the range considered to represent the closest residential location with continuous occupancy. The distances of 400 m and 700 m were found to be representative of the highest dose locations for releases into the building wake and stack releases respectively, for the 'D5' weather condition.

The wind direction was assumed to remain unchanged for the duration of the release. The exposed individual is assumed to remain outdoors and directly downwind for the duration of the release and then remain at the same location following the release, returning to a representative occupancy pattern of 10 percent outdoors, 90 percent indoors following the end of the release. An additional location factor of 0.1 was applied to account for shielding provided by building structures against long term ground shine.

It is noted that food ingestion is not included as a pathway for allocation to facility dose bands as the release categories representing damaged core conditions would be expected to result in implementation of food restrictions (i.e. the European Council Food Intervention Levels, CFILs) [Ref-25.71]. This would prevent this pathway from becoming a significant contributor to the predicted public doses close to the site.

It is noted that some of the habit assumptions used for the deterministic dose calculations are quite conservative. For example, the individual at the most exposed location is assumed to remain outside throughout the duration of the release, even where the release is prolonged, and then remain at this exposure location continuously for the long term. However, such assumptions have been retained for simplicity.

25.6.3.3 Dose Coefficients

The overall approach for calculating the deterministic dose to representative members of the public involves the application of dose coefficients to TIAC values of radionuclides in the environment or committed dose coefficients in the body following an intake. In the current assessment, the dose coefficients used to calculate doses due to internal exposure were taken from ICRP Publication 119 [Ref-25.68]. These dose coefficients give the committed dose to members of the public integrated over a period of 50 years (for adults) or up to the age of 70 years (for children and infants), following the intake of radioactivity.

For inhalation pathways, the lung clearance type for radionuclides, which corresponds to how readily material is absorbed into the blood from the respiratory tract, is selected on the basis of guidance given in

ICRP-71 [Ref-25.60]; otherwise the medium clearance type (or the higher value, where only two clearance types are available) is assumed.

Age specific breathing rates are taken from ICRP-71 [Ref-25.60].

For external pathways, the methodology used to calculate the cloud gamma dose factors is based on the semi-infinite cloud model described in HPA RPD-058 [Ref-25.69]. The photon energies and yields required for each radionuclide were taken from ICRP-107 [Ref-25.70]. For the gamma dose due to radioactivity deposited on the ground, dose per unit deposit coefficients were taken from HPA-RPD-032 [Ref-25.61] and supplemented with coefficients based on Table III.3 of FGR12 [Ref-25.62].

25.6.4 Probabilistic Risk Calculations

Probabilistic consequence calculations, for assessment against the individual risk target (Target 7) and societal risk target (Target 9), were performed using the accident consequences code PC COSYMA version 2.03. This version of the code has been modified specifically for UK conditions [Ref-25.71] and was selected on the basis that it is the only code currently commercially available that is capable of calculating the required probabilistic risk of health effects using models and data appropriate to the UK.

The key model parameters, assumptions and inputs are summarised below.

25.6.4.1 Atmospheric Dispersion Methodology

PC COSYMA incorporates the MUSEMET model, a segmented Gaussian plume model that allows temporal changes in the meteorological conditions affecting the dispersing material, i.e. hourly changes in the wind speed and wind direction, stability category and rainfall rate. For long duration releases, the model can be used with a phased approach, in which releases are divided into a series of phases (up to 6 representative hourly phases). All material released in a particular phase follows the same trajectory, but material released in different phases can follow different trajectories depending on wind direction at the start of a phase.

For the current Level 3 PSA study, weather conditions for the Wylfa Newydd site are considered using UK Meteorological Office's Numerical Weather Prediction (NWP) datasets for the ten years period 2003 to 2013. The meteorological datasets include hourly sequential data on the wind direction, wind speed, atmospheric stability, rainfall and mixing layer depth. Due to limitations in the amount of hourly weather data that can be processed by PC COSYMA, the 2012 dataset was selected for use in the assessment (a prior review had shown this dataset to be representative of the 10 years period). A cyclic weather sampling scheme was used to sample the 2012 meteorological data, generating 144 weather sequences of 61 hours interval and an 8 hours shift relative to the start time in the dataset was applied.

The same classes of material are used in the dispersion calculations for the probabilistic risk assessments as were used for the deterministic dose assessments, i.e. noble gases, aerosols and iodine in three forms of CsI, I_2 and CH_3I).

25.6.4.2 Exposure Pathways and Habit Assumptions

Following radionuclide release to atmosphere, a representative (adult) member of the public may be exposed to radioactivity in air or deposited on the ground via a number of pathways. The dose uptake pathways considered in PC COSYMA include:

- External gamma irradiation by radioactive material entrained in the passing plume (often referred to as cloud gamma or cloud shine),
- Internal irradiation due to inhalation of radioactive material entrained in the passing plume,

- External beta irradiation by material deposited on skin or clothing,
- External gamma irradiation from material deposited on the ground,
- Internal irradiation following inhalation of material re-suspended after its initial deposition on the ground (long term), and
- Internal irradiation following the ingestion of foodstuffs contaminated by deposited material.

For both the individual and societal risk calculations, no offsite protective actions are considered in the basic calculations beyond limiting ingestion dose in accordance with legally mandated limits of radioactivity in foodstuffs [Ref-25.72].

The granularity in predictions of individual risk is limited by the distance bands in the PC COSYMA population grids. Conditional individual risks were calculated at distances of 400m (midpoint of 0 to 800 m), 1,000 m (800 m to 1,200 m), and 1,500 m (1,200 m to 1,800 m) from the Site. This is done to illustrate how the individual risk varies with distance over the range of interest. The distance of 1,000m was used as the main reference point for the assessment against Target 7. Predictions of societal risk (short term fatalities and notional late fatalities in the UK population) were based on the UK population file, compiled in the form of a radial grid centred on the Wylfa Newydd site.

25.6.4.3 Dose and Risk Coefficients

PC COSYMA incorporates data libraries for many of the quantities required to perform the calculations. The data libraries comprise dose and risk coefficients, as well as population and agricultural gridded data centred on the site. These default library data were used for the current study.

It is noted that although the dose coefficients are for adults, the risk coefficients were adjusted by HPA to account for the UK population as a whole [Ref-25.71].

25.6.4.4 Health effects considered in the risk calculations

The endpoints of particular interest in the current probabilistic calculations are:

- The mean individual risks of early and late fatalities (over all sectors and weather sequences) as a function of distance from the site, and
- The mean number of early and late fatalities (over all weather sequences) in the affected population.

Calculation of conditional short term individual risk and the number of early fatal health effects uses models for the following deterministic health effects:

- Pulmonary syndrome,
- Haematopoietic syndrome,
- Gastrointestinal syndrome,
- Pre- and neonatal death, and
- Skin burns leading to death.

Estimation of long term individual risk and number of notional late fatalities considers fatal cancers, including leukaemia, for a number of organs (bone marrow, bone surface, breast, colon, liver, lung, pancreas, skin, stomach, thyroid and remainder) and hereditary effects.

25.7 Internal Events at Power PSA (Results and Insights)

25.7.1 Introduction

This section provides the results and insights for internal events at power PSA. The results and insights provided in this section cover internal events at power level 1, 2 and 3 PSA including sensitivity and uncertainty analysis. The results and insights on shutdown PSA, spent fuel pool PSA, PSA for internal/external hazards, fuel route PSA and non-reactor fault PSA are provided in later sections.

25.7.2 Results of Level 1 PSA

The purpose of this section is to document the calculation of the Core Damage Frequency (CDF) due to events that occur when the plant is operating at power. This section covers internal events at power only. The calculation of CDF for external events and shutdown conditions is described in other sections.

The calculation of the IEAP CDF is performed as a single top gate. The use of the sequence markers results in some non-minimal cutsets; however, this contribution is small and is judged not to significantly impact the insights.

Section 25.4 presents the quantification methodology.

25.7.2.1 Core Damage Frequency

The IEAP CDF which satisfies the criterion of convergence (see Section 25.7.2.1(1)) is calculated to be 2.27E-07 /y.

(1) Model Convergence

The model was quantified at different truncation levels to evaluate convergence. The truncation analyses were performed from 1E-09 to 1E-17 using sequence by sequence quantification (and as necessary IE by IE quantification in each sequence).

According to the results shown below, the convergence is considered sufficient since successive reductions in truncation value of one decade result in the decreasing changes of CDF and the final step change is less than 5 percent.

From the next Section ((2) Initiating Event Contribution), the results are presented based on the single-top quantification with the truncation value of 1E-14.

(2) Initiating Event Contribution

Table 25.7.2-1 shows a summary of contributions to CDF in the form of a pie chart.

Contribution by broad IE categories

Transients (including LOOP):	46 %
Manual shutdowns (including support system initiators):	39 %
LOCAs (including BOC/ISLOCA):	14 %

Transients have the highest contribution among these broad IE categories. This result is consistent with the other plant PSAs [Ref-25.2] [Ref-25.39].

The contribution to CDF from the manual shutdowns is comparable with that from the transients. The manual shutdowns contribution is much smaller than the CDF from transients in other plant PSAs [Ref-25.2] [Ref-25.39]. This is because the support system initiators and Tech. Spec. initiators are systematically identified and modelled in the UK ABWR PSA, including CCF events.

The smallest contribution is from LOCAs due to the smaller IE frequencies. The IE frequencies for LOCA are three orders of magnitude less than for transients.

Transients

General transient (TG)	6 % of IEAP CDF
Loss of condenser heat sink (TM)	7 % of IEAP CDF
Loss of feedwater (TF)	6 % of IEAP CDF
Inadvertent open relief valve (TI)	< 0.01 % of IEAP CDF
Inadvertent ADS actuation (A-ADS)	< 1 % of IEAP CDF
PCV pressurisation by leakage of nitrogen gas (TP)	< 0.1 % of IEAP CDF
Loss of offsite power (TE)	28 % of IEAP CD
< 0.5 h (TE1)	2 % of IEAP CDF
Plant centred (TEP1)	< 1 % of IEAP CDF
Switchyard centred (TES1)	1 % of IEAP CDF
Grid related (TEG1)	< 1 % of IEAP CDF
Weather related (TEW1)	< 1 % of IEAP CDF
0.5 to 8 h (TE2)	16 % of IEAP CDF
Plant centred (TEP2)	< 1 % of IEAP CDF
Switchyard centred (TES2)	5 % of IEAP CDF
Grid related (TEG2)	9 % of IEAP CDF
Weather related (TEW2)	1 % of IEAP CDF
8 to 14 h (TE3)	4 % of IEAP CDF
Plant centred (TEP3)	< 0.1 % of IEAP CDF
Switchyard centred (TES3)	1 % of IEAP CDF
Grid related (TEG3)	2 % of IEAP CDF
Weather related (TEW3)	< 1 % of IEAP CDF
> 14 h (TE4)	5 % of IEAP CDF
Plant centred (TEP4)	< 0.1 % of IEAP CDF
Switchyard centred (TES4)	1 % of IEAP CDF
Grid related (TEG4)	2 % of IEAP CDF
Weather related (TEW4)	3 % of IEAP CDF

The highest contributor among the transients is Grid related LOOP for 0.5 to 8 hours (TEG2) (sequences 7, 8 and 9) (see Section 25.7.2.1(4)).

The second highest contributor among the transients is Loss of condenser heat sink (TM) (sequence 2) (see Section 25.7.2.1(4)).

The third highest contributor among the transients is General transient (TG).

The fourth highest contributor among the transients is Loss of feedwater (TF) (sequence 4) (see Section 25.7.2.1(4)).

Other transients, i.e. Inadvertent open relief valve (TI), Inadvertent ADS actuation (A-ADS), PCV pressurisation by leakage of nitrogen gas (TP), have less than 1 percent contribution to the IEAP CDF.

Manual shutdowns

Manual Shutdown (MS) contributes approximately 39 percent to the IEAP CDF. This group includes manual shutdowns required by Technical Specifications as well as manual shutdowns due to loss of support systems which do not cause an immediate plant scram. Therefore, MS also involves some degradation of the front-line systems required to function for safe shutdown after plant shutdown.

The most significant IE among the manual shutdowns is Loss of RCW/RSW (IE-RCW_RSW) and this is the highest among all the IEs. This IE includes loss of two or more divisions of RCW/RSW. Loss of all RCW/RSW divisions, dominated by CCF of all RCW pumps, RSW pumps or RSW strainers, have a wide range of dependent failures, e.g. all HPCF/RHR divisions, feedwater system, condensate system, main condenser, and EDGs.

The second most significant IE among the manual shutdowns is Loss of HNCW (IE-HNCW) and this is the 5th highest IE. This IE causes loss of feedwater system, condensate system and main condenser (not immediate losses but gradual degradation). In addition, this IE causes un-recoverable LOOP given the outdoor temperature is above 25 ^{o}C.

The third most significant IE among the manual shutdowns is Loss of TBNEE/Z HVAC (IE-HVTE) and this is the 8th highest IE. This IE causes un-recoverable LOOP.

The most significant sequence for manual shutdowns is MS-HCTL12 (sequence 1) (see Section 25.7.2.1(4)).

LOCAs

The most significant LOCA initiator is RPV rupture (S4), labelled as “Excessive LOCA”. This IE is assumed to be un-mitigatable and results directly in low pressure core damage. It is notable that this IE frequency has large uncertainty (EF=62) as shown in Table 25.4.1-2. Also, the modelling uncertainty about the accident sequence is discussed in Topic Report on IEAP Level 1 PSA [Ref-25.9].

The second highest contributor among LOCAs is BOC (Breaks Outside Containment) at the main steam line (including upstream and downstream of outboard MSIVs). Due to the relatively high break frequency, relatively high probability of isolation failure (dominated by software failure for MSIV) and relatively less mitigation systems credited, the CDF from this IE is the highest among the BOC events.

The ISLOCA of the RHR suction lines involve CCF spurious opening of two isolation MOVs. Isolation by the operator is not credited. The ISLOCA of the RHR suction lines is modelled as large LOCA and so FLSS is insufficient. These are the major reasons for the higher contribution from the RHR suction line ISLOCA than the HPCF/LPFL injection line ISLOCAs.

(3) Accident Class Contribution

An additional perspective on the Level 1 at Power PSA results is provided by examining the relative contributions to the CDF by the accident classes used to define the Level 1 end states of the event trees. Figure 25.7.2-2 illustrates the CDF contribution of each accident class in the form of a pie chart. A description of each of the end state codes is presented in Table 25.4.3-2.

The accident classes each contributing 1 percent or more of the IEAP CDF is discussed.

- The significantly largest accident class contributor is TQUV (62 percent), which involves a non-LOCA event with failure of high pressure and low pressure injection, but successful depressurisation, resulting in low pressure core damage in the short term (includes sequences 1, 4, 5, 7, 9 and 10).
- The second largest contributing accident class is S3E (8 percent). This class involves ISLOCAs or BOCs with failure of RPV injection, resulting in low pressure core damage in the short term with containment bypass. ISLOCA/BOC with failure of reactivity control is also grouped as S3E since a similar end state would be expected (includes sequence 6).
- The third largest contributing accident class is TQUX (8 percent). This class involves a non-LOCA event with failure of high pressure injection and RPV depressurization, resulting in high pressure core damage in the short term (includes sequence 8). Note that sequence 8 represents TE221 in which core damage occurs after operation of RCIC for 8 hours, which is not actually “short term” but is bounded by TQUX.
- The fourth largest accident class contributor is AC (7 percent), which involves a LOCA inside containment (including consequential LOCAs) with failure of reactivity control (ATWS), resulting in containment failure in the short term followed by core damage at low pressure (includes sequence 2). ATWS followed by reactor coolant pressure boundary failure (due to consequential LOCA, power-flow oscillation, RIP trip failure, etc.) is also categorised as AC and this dominates the AC frequency rather than the LOCA followed by ATWS.
- The fifth largest contributing accident class is S4 (5 percent). This class involves an RPV rupture (excessive LOCA inside containment), directly resulting in low pressure core damage in the short term. RPV rupture is assumed to be un-mitigatable. It is notable that this IE frequency has large uncertainty as shown in Table 25.4.1-2.
- The sixth largest frequency contributing accident class is TB (3 percent), which involves a loss of Class 1 and Class 2 AC power resulting in high pressure core damage after 8 hours of RCIC operation.
- The seventh largest frequency contributing accident class is TBU (3 percent), which involves loss of Class 1 and Class 2 AC power with failure of RCIC, resulting in high pressure core damage in the short term.
- The eighth largest frequency contributing accident class is TBP (1 percent), which involves loss of Class 1 and Class 2 AC power with a stuck open relief valve (causing loss of RCIC), resulting in low pressure core damage in the short term. This is similar to TB, except that RCIC does not operate due to the stuck open relief valve, which changes the timing of core damage.
- The ninth largest frequency contributing accident class is AE (1 percent), which involves loss of injection at LOCAs inside containment (including Consequential LOCA), resulting in low pressure core damage in the short term.

(4) Accident Sequences

Seventy-seven sequences match the definition of significant accident sequences (definition provided in Section 25.4.8.1(2)). Figure 25.7.2-3 illustrates the CDF contributions by the accident sequences in the form of a pie chart.

The top ten accident sequences which contribute to CDF are described below.

Rank 1. MS-HCTL12

This sequence is a manual shutdown with failure to maintain power conversion systems and failure of high pressure injection by feedwater pumps. MSIV closure occurs as designed at the low water level L1.5 which causes SRVs to open. The SRVs successfully reclose. RCIC successfully injects, but S/P cooling by RHR fails. The event tree transfers from MS to MS-HCTL. The operator successfully depressurises the RPV, but injection by condensate pumps, FLSS and FLSR (when FLSR is credited) fails. Even though this sequence is not "short term", the end state is assigned to short term core damage at low pressure (TQUV).

The high contribution is the result of:

- Manual shutdown IEs share common event trees (MS, MS-HCTL), so that contributions from multiple IEs are combined; and
- CCF events causing loss of all RCW/RSW are considered as the failure modes causing IEs. Loss of all RCW/RSW disables all the injection systems except RCIC (for initial 8 hours only), FLSS and FLSR. The cutsets including such failure modes have high contributions to this accident sequence and to the IEAP CDF.

Rank 2. TM-A22

This sequence is Loss of condenser heat sink followed by failures of RPS scram, ARI and RPT. Failure of RPT during an ATWS results in a consequential LOCA. Since SLC is not credited given LOCA, containment overpressure failure occurs due to a continuous generation of a large amount of steam. Core damage occurs due to loss of ECCS. The end state is short term core damage at low pressure due to the combination of LOCA and failure of reactivity control (AC).

The reasons for the high contribution among the Class AC are as follows.

- ARI and RPT credit initiation via only the high reactor pressure signal following Loss of condenser heat sink (TM) while more than one signal is credited in other transients and RPT is not required in LOOP IEs. That is why TM-A22 has a significantly higher frequency than TG-A22, TF-A21 and TI-A12.
- ARI and RPT depend on common instrumentations, such that CCF events fail both ARI and RPT. That is why TM-A22 has significantly higher frequency than TM20 in which failure to open SRVs is the cause of consequential LOCA.
- The Feedwater stop function also depends on the same instrumentation (high RPV pressure) as ARI and RPT. However, consequential LOCA and core damage caused by failure to stop feedwater (TM21) has much lower frequency than TM22 because the heading FDW-STOP is located after RPT-P-A. Therefore, the minimal cutsets including CCF of high reactor pressure instrumentations are assigned to TM-A22 rather than TM-A21.

Rank 3. S401

This sequence is RPV rupture, directly resulting in core damage. Seven or more W/W to D/W vacuum breakers successfully open after the initial pressurization transient, so the containment is intact as the Level 1 PSA end state (Class: S4E). There is only one cutset for this sequence but this is the highest cutset.

It is notable that this IE frequency has large uncertainty.

Rank 4. MS-HCTL41

This sequence is a manual shutdown with failure to maintain both the power conversion systems and high pressure injection by the feedwater pumps. MSIV closure occurs successfully at the low water level L1.5 which causes the SRVs to open. The SRVs successfully reclose. RCIC successfully injects, but S/P cooling by RHR fails. The event tree transfers from MS to MS-HCTL. The operator fails to depressurise the RPV. RCIC becomes inoperable at 8 hours due to the increased S/P temperature (by the exhaust steam from the RCIC turbine). After that, high pressure injection by HPCF fails. Automatic depressurisation by Transient ADS and RDCF is successful, but all the low-pressure injection systems fail. Even though this sequence is not "short term", the end state is assigned to short term core damage at low pressure (TQUV).

Rank 5. TF-HCTL09

This sequence is similar to the highest contributing sequence (MS-HCTL12) except for the nature of the IE. The power conversion system, feedwater system and condensate system are not examined due to the known (non-probabilistic) dependency. Therefore, the CDF is higher than similar sequences for other transient IEs, e.g. TG-HCTL12, TM-HCTL11.

Rank 6. S3-BMSFB02

This sequence is BOC of a main steam line or feedwater-B line, followed by failure of isolation, success of reactivity control, no conditional LOOP, and success of HPCF or LPFL. Since the water or steam leaking from the RPV does not return to S/P, HPCF and LPFL are credited only until the loss of their NPSH in S/P (NOTE: makeup to CST from PWST to extend HPCF operation is not credited in these IEs). Therefore, switching of the injection system to FLSS after loss of ECCS is required but it fails in this sequence. The end state is core damage at low pressure with containment bypass (S3E).

The reasons of the high contribution among the Class S3E are discussed.

- The IE "BOC at MS line" has relatively high break frequency among the BOC events because this IE includes the MS drain line (small diameter and high break frequency).
- Failure of all FLSSs occurs with higher probability than failure of all ECCS or failure of control rod insertion, such that this sequence has higher frequency than other sequences in the same event tree, e.g. S3-BMSFB03, S3-BMSFB07.

Rank 7. TE220

LOOP occurs as the IE. Reactivity control and opening and closing SRVs are successful. All EDGs fail. RCIC is successfully initiated. At 8 hours, offsite power is recovered but containment heat removal by RHR fails, which causes loss of RCIC due to high S/P temperature. After the loss of RCIC, RPV is depressurised by operator or RDCF signal but low pressure injection by FLSS and FLSR both fails. Even though this sequence is not "short term", the end state is assigned to short term core damage at low pressure (TQUV).

Although recovery of offsite power in 8 hours is assumed for this IE, HPCF and LPFL are not credited after the loss of RCIC because the dominant failure modes impacting RHR (e.g. Software CCF of HVAC, RHR_RCW) also disable LPFL and HPCF.

Rank 8. TE221

This sequence is similar to TE220. The difference to TE220 is failure of RPV depressurisation. Even though this sequence is not "short term", the end state is assigned to short term core damage at high pressure (TQUX).

The reasons for the high contribution among the Class TQUX are as follows.

- Occurrence of this sequence requires a relatively small number of failures: EDGs, RHR and RPV depressurisation. EDGs and RHR have many common cause failure modes, e.g. Software failure of HVAC, RHR_RCW.
- Only RDCF is credited for RPV depressurisation. More SRVs are required to open for RDCF compared to those for Transient ADS.

Rank 9. TE2-HCTL09

This sequence is similar to the highest contributing sequence (MS-HCTL12). The only difference is the nature of the IE. The power conversion system, feedwater system and condensate system are not examined due to the known (non-probabilistic) dependency.

Rank 10. MS20

This sequence is a manual shutdown with failure to maintain power conversion systems and also high pressure injection by feedwater pumps. MSIV closure occurs successfully at the low water level L1.5 which causes SRVs to open. The SRVs successfully reclose. High pressure injection by RCIC and HPCF fails. RPV depressurisation succeeds but all the low-pressure injection systems (LPFL, CS, FLSS) fail. The end state is short term core damage at low pressure (TQUV).

This sequence has the highest CDF among those for TQUV in actual “short term” (not HCTL scenarios). That is because manual shutdown IEs share the common event tree (MS), so that contributions from multiple IEs are combined. This sequence has significantly smaller frequency than the highest contributing sequence (MS-HCTL12) because the failure modes related to RCIC are also included in the cutsets.

(5) Importance analysis

This section identifies the top contributors to the CDF in various categories of events based on the Fussell-Vesely (F-V) risk importance measure and, when insights can be realised, the Risk Achievement Worth (RAW) measure is examined. In each case, the basis for the significance is discussed.

Components important to safety (high F-V events)

The components important to safety are discussed here in terms of their F-V values. Table 25.7.2-1 shows the importance of events sorted by F-V in descending order. In this subsection, the initiating events, CCF events, test and maintenance events, human failure events and other marker events (e.g. IE markers, sequence markers, class markers, etc.) are excluded from the discussion.

FLSR

FLSR-UNAVAILABLE “FLSR (Mobile Injection Facility) Unavailability” has a F-V value of 0.22. This high F-V is due to the appearance of FLSR failure in many significant sequences, e.g. HCTL condition, Loss of Class 1 and Class 2 AC power (station blackout). Also, FLSR failure (including mechanical failure and human error) is simply represented in the internal events at power PSA as a single basic event (i.e. a ‘super-component’) with a conservative probability of failure. This will be addressed further during the plant-specific stage.

Backup Building Generators

BBG-1 “Loss of BBG-1” has a F-V value of 0.14. BBG-2 “Loss of BBG-2” has a F-V value of 0.13. These high F-V values are due to many significant sequences involving loss of Class 1 AC power, e.g. LOOP with failure of all EDGs.

Although the two events have the same failure probability and there is the same symmetrical system structure of FLSS divisions 1 and 2 including support systems, the F-V value of BBG-2 is slightly lower than that of BBG-1. This slight difference occurs because the hardened containment venting system is supported by BBG-1 while the filtered containment venting system is supported by BBG-2. The filtered containment venting system includes COPS which does not need AC power.

Emergency Diesel Generators

R43-DG_-RA-__-D/G-A "D/G-A Fail to Run > 1 Hour" has a F-V value of 0.039, R43-DG_-RA-__-D/G-B "D/G-B Fail to Run > 1 Hour" has a F-V value of 0.067, and R43-DG_-RA-__-D/G-C "D/G-C Fail to Run > 1 Hour" has a F-V value of 0.072. Due to the high frequency of LOOP events (including some support system initiators) and high conditional LOOP probability, EDGs are important.

RCIC pump

E51-TPR-FS-__-C001 "Turbine-Driven Pump (RCIC) C001 Fail to Start" has a F-V value of 0.063. The RCIC pump is obviously important because RCIC does not depend on AC power, RCW and room cooling (assumed), which provides diversity against HPCF and LPFL.

SRV (reclosing)

SRV-C "Failure of SRV reclosure" has a F-V value of 0.059. This failure mode disables RCIC, PCS and some operator actions (due to reduced time available).

Class 3 AC bus (M/C) 1A1

R22-BSA-LF-__-M/C-1A1 "Bus (AC Power) M/C-1A1 Loss of Function" has a F-V value of 0.027. The other three buses (1A2, 1B1, 1B2) have F-V values of less than 0.001.

The major reasons of the significantly higher F-V of 1A1 than those of other Class 3 buses are:

Loss of M/C-1A1 causes loss of two HNCW pumps / chillers resulting in loss of HNCW while loss of each of other Class 3 M/Cs causes loss of one HNCW pump / chiller which does not result in loss of HNCW as far as other four pumps / chillers are functioning.

M/C-1A1 supplies AC power to two buses (Class 1 M/C-C and Class 2 P/C-BB1) while each of the other Class 3 M/Cs supplies power to one bus (Class 1 or Class 2).

CST

CST-INVENTORY "Failure probability for inadequate inventory in the CST" has a F-V value of 0.027. This failure mode disables RCIC, HPCF, the feedwater system and the condensate system.

Test and Maintenance events

Among the T&M events, the top ten events in terms of F-V are listed in Table 25.7.2-2.

Rank 1, 2:

Events representing T&M of FLSS-B and its room cooling (HVAC-BB-B) have the same probability and impact, which results in equivalent F-V values of 0.06. The high F-V value corresponds to the highest contribution of TQUV where FLSS fails.

Rank 3, 4:

Events representing T&M of FLSS-A and its room cooling (HVAC-BB-A) are similar to the top two ranking events. The reason for the slightly lower F-V value for A train compared to train B is the slightly higher importance of loss of BBG-1 compared to loss of BBG-2 (see Section 25.7.2.1 (5) (1)). It can be seen that T&M of FLSS-B (or HVAC FLSS-BB-B) appears with loss of BBG-1 in many cutsets while T&M of FLSS-A (or HVAC FLSS-BB-A) appears with loss of BBG-2 in many cutsets.

Rank 5:

This event represents T&M of the FLSS / FLSR common injection line. This corresponds to the highest contributor to TQUV among the Accident Classes. Although this event impacts both FLSS trains and FLSR, the F-V value is comparable with the Rank 2 and Rank 4 events. This is because the unavailability of this event is two orders of magnitude smaller.

Rank 6:

The event representing T&M of RHR-A has a higher F-V value than those representing T&M of RHR-B and RHR-C. This is due to the slightly lower F-V value of EDG-A compared to EDG-B and EDG-C (see Section 25.7.2.1 (5) (1)). It can be seen that T&M of RHR-A appears with loss of EDG-B and/or EDG-C in many cutsets.

Rank 7,8:

These events represent T&M of BBG-A and BBG-B. As discussed in Section 25.7.2.1 (5) (1), BBG is important. The reason for the slightly higher F-V value of BBG-1 compared to BBG-2 is explained in Section 25.7.2.1 (5) (1).

Rank 9:

The event representing T&M of EDG-C has a slightly higher F-V value than the events representing T&M of EDG-A and EDG-B. This reflects the order of importance of the EDGs as discussed in Section 25.7.2.1 (5) (1). The lower F-V value of T&M of EDG-B could be explained by the existence of hardwired backup control for Division III systems, e.g. HPCF, RSW and HVAC.

Rank 10:

The event representing T&M of EDG-B has a slightly lower F-V value than that representing T&M of EDG-C and a higher F-V value than that representing T&M of EDG-A.

It was noted that the top ten F-V T&M events also appear as the top ten RAW T&M events.

Post-Initiator HFEs

Among the post-initiator HFE events, the top ten events in terms of F-V are listed in Table 25.7.2-3.

Rank 1:

Since FLSR-UNAVAILABLE is considered to be dominated by human error, this event is included in this section. This event has the highest F-V value among all the post-initiator HFEs.

Rank 2:

HCTL-based ignition of RPV depressurization and initiation of FLSS operator action is required at high frequency conditions, e.g. transients or manual shutdowns followed by success of RCIC or HPCF.

Rank 3:

Failure of manual RPV depressurisation as the backup of Transient ADS has a high F-V value because many of the TQUX sequences (Rank 2 Class) include this HFE.

None of the other post-initiator HFEs satisfy the criteria of a significant basic event in terms of F-V value (i.e. above 0.005).

Pre-Initiator HFEs

Among the pre-initiator HFE events, the top ten events in terms of F-V are listed in Table 25.7.2-4. It can be seen that no pre-initiator HFEs satisfy the criteria of a significant basic event in terms of F-V value (i.e. above 0.005). Therefore, the top ten events in terms of RAW are listed in Table 25.7.2-5 and discussed below.

Rank 1:

CCF Miscalibration of Pressure Transmitter B21-PT009ABCD has the highest RAW. It disables the high reactor pressure signal to ARI, RPT, SLC and the Feedwater stop function. It is noted that only initiation via the high reactor pressure signal is credited for the ATWS systems for the Loss of condenser heat sink event and LOOP events.

Rank 2:

CCF Miscalibration of Level Transmitter B21-LT-FLSS-ABCD has the second highest RAW. It disables automatic RPV depressurisation by RDCF and automatic injection by FLSS, which are credited for a wide range of IEs / sequences.

Rank 3:

CCF Miscalibration of Pressure Transmitter B21-PT-FLSS-ABCD has the third highest RAW. It disables the automatic opening of the FLSS injection valve even if FLSS pumps are successfully initiated. The RAW is slightly lower than that of the Rank 2 event because this failure mode does not disable RDCF while the Rank 2 event disables both RDCF and FLSS.

Rank 4:

CCF Miscalibration of Level Transmitter B21-LT003JKLM has the fourth highest RAW. It disables low water level (L2) signal to ARI, RPT and SLC. The RAW is lower than that of the Rank 1 event because the frequencies of IEs that depend on only low water level ATWS signal (i.e. IORV, LOCAs) are smaller than the frequencies of IEs which depend on only high reactor pressure ATWS signal (Loss of condenser heat sink, LOOPs).

Rank 5:

Manual valve F005B left closed after maintenance (outage) disables FLSS-B. Due to the importance of FLSS-B, this event has high importance.

Rank 6:

Manual valve F005A left closed after maintenance (outage) disables FLSS-A. Due to the importance of FLSS-A, this event has high importance.

As discussed in Section 25.7.2.1 (5) (1), BBG-1 has a slightly higher F-V value than BBG-2. Manual valve F005A left closed appears with loss of BBG-2 in many cutsets while Manual valve F005B left closed appears with loss of BBG-1 in many cutsets. That is the reason why the RAW value of F005A is slightly lower than that of F005B.

Rank 7

CCF Miscalibration of Pressure Transmitter B21-PT007ABCD (Higher drift) has the seventh highest RAW. This event disables all LPFLs due to the loss of permission signal to open their injection valves.

Rank 8

Four events regarding restoration error of the Backup Building DC circuit breakers are ranked. Each of them disables ARI.

System Level Importance

All the basic events that have the unique system ID (including pre-initiator HFEs) are assigned to one of the “systems” specified by the system ID. Some miscellaneous basic events, e.g. BBG-1, BBG-2, FLSR unavailability and VSS related events are manually checked for specifying the relevant systems. The CCF events are included for the purpose of calculating the system level F-V. The post initiator HFEs are not included. The IE basic events (having own “frequency”) are also not included. The sequence flag events and the accident class flag events are also not included.

The system level F-V value of a system is calculated by setting all the relevant basic events including CCF events to FALSE in the cutset file based on the 1E-14 truncation.

Table 25.7.2-6 and Table 25.7.2-7 show the system level importance in terms of F-V and RAW respectively.

The observations are as below.

F-V:

- CONTROL PANEL (digital C&I components and software) has the highest F-V value. The reasons for the high F-V are the wide use of digital C&I system for Class 1 and Class 3 safety systems and the conservative modelling of software CCF in terms of probability and consequence.
- FLSS has the second highest F-V value. This is because FLSS is credited for most of the IEs and because most of the TQUV sequences involve failure of FLSS. Also, FLSS is credited to prevent core damage even after containment failure.
- BBG has the third highest F-V value. The reason for the high F-V value is that BBG is necessary to operate FLSS and manual containment venting in the LOOP conditions (including conditional LOOP and LOOP caused by support system initiators) which occurs in high frequency.
- EDG has the fourth highest F-V value. The reason for the high F-V value is that the EDGs are needed to operate HPCF, RHR and their support systems in the LOOP conditions (including conditional LOOP and LOOP caused by support system initiators) which occurs at high frequency.
- FLSR has the fifth highest F-V value. FLSR has high unavailability and it is credited for risk significant sequences, e.g. SBO, loss of all RCS/RSW.

- SRV has the sixth highest F-V value. Various failure modes of SRVs are modelled: failure to open for depressurisation, failure to open for pressure boundary protection, failure to reclose causing SORV.
- NOTE: SRV is part of the Nuclear Boiler system. However, the Nuclear Boiler System is separately assessed due to its special role in safety.
- HVAC has the seventh highest F-V value. The reason for the high F-V value is that room cooling is needed for a wide range of systems/components, e.g. HPCF, RHR, FLSS, EDG, BBG, RCW, RSW, TCW, TSW, FW, CS, PCS, buses, controllers.
- Nuclear Boiler system (NB), excluding SRVs, has the eighth highest F-V value. Various SSCs are involved: reactor vessel instrumentation, MSIVs, FW-A line check valves (forms injection line of RHR-A, FLSS, FLSR), etc.
- RSW has the ninth highest F-V value. Loss of RSW impacts a wide range of systems. RSW has a higher F-V value than RCW because RSW includes unique failures modes compared to RCW, e.g. strainer plug.
- RCIC has the tenth highest F-V value. RCIC is still available for 8 hours given loss of HVAC, RCW, RSW and Class 1 AC power.

RAW:

- Nuclear Boiler system (NB), excluding SRVs, has the highest RAW value. Various SSCs are modelled: reactor vessel instrumentation, MSIVs, FW-A line check valves (forms injection line of RHR-A, FLSS, FLSR), etc.
- RHR has the second highest RAW value. All the HCTL sequences, which significantly contribute to the IEAP CDF, involve failure of the RHR heading.
- CONTROL PANEL (digital C&I components and software) has the third highest RAW value. The reason for the high RAW value is the wide use of the digital C&I system for Class 1 and Class 3 safety systems.
- DC has the fourth highest RAW value. Loss of DC power fails all the injection systems except FLSR.
- M/C has the fifth highest RAW value. Most of the injection systems except RCIC (for 8 hours) and FLSR depend on AC power.
- HVAC has the sixth highest RAW value. Loss of HVAC for the heat exchanger building causes the RCW/RSW, and then loss of all RCW/RSW disables most of the frontline systems except reactivity control systems, RCIC (for 8 hours), FLSS, FLSR, containment venting system, ADS, RDCF.
- P/C has the seventh highest RAW value. Most of the injection systems except RCIC (for 8 hours) and FLSR depend on AC power.
- MCC has the eighth highest RAW value. Most of the injection systems except RCIC (for 8 hours) and FLSR depend on AC power.
- HPCF has the ninth highest RAW value. Most of the core damage sequences include failure of HPCF.
- FLSS has the tenth highest RAW value. FLSS is credited as mitigation for most of the IEs.

(6) PSA Insights

The quantification results provide the following insights:

- Transients have a large contribution (46 percent) to the total CDF. Two third of the CDF by transients comes from LOOP events because restart of important support systems, e.g., RCW, HVAC, is needed and this depends on digital control system. LOOP with failure of digital control system (Software CCFs, CPU CCFs) results in SBO for the Class 1 systems.
- Manual shutdowns (including support system initiators) have a considerable contribution (39 percent) to the total CDF. More than one third of the CDF by manual shutdowns comes from loss of RCW/RSW because CCFs of all the RCW/RSW divisions are included.
- About one third of the CDF by LOCAs comes from RPV rupture. Another one third comes from BOC events. The remaining one third comes from LOCAs inside containment and ISLOCAs.
- TQUV is the most significant Accident Class: about 60 percent of the total CDF. This accident class includes various scenarios that are not short-term are included, e.g., failure of low pressure injection after operation of RCIC or HPCF has operated for 3, 8 hours or more.
- FLSS and BBG have F-V values of more than 50 percent and 40 percent (system level), respectively. This is a notable insight and may raise the needs for improving the reliability and adequate test and maintenance. The detail design on these systems is not determined in GDA and thus these systems are modelled as the super component in the relevant fault tree.
- EDG also has high system level importance (nearly 40 percent) similar to BBG. If Station Blackout occurs, RCIC is the only mitigation system to inject water into RPV in short term. Therefore, prevention of Station Blackout is very important for UK ABWR.
- HVAC has a F-V value of more than 30 percent because of the wide range of impacts.
- FLSR is credited primarily for the severe accident mitigation. However, this also contributes significantly to the accident mitigation before severe accident.
- Digital C&I system has more than 40 percent F-V (system level). This implies the importance of digital C&I system reliability. But this should be noted that conservative software CCF significantly contributes to that high F-V.
- Software CCFs (HVAC, RPS and RHR_RCW) have large contributions to the total CDF because the CCF probability is conservative.
- Test and maintenance unavailabilities of less redundant systems, e.g., FLSS, BBG have higher importance than the more redundant systems, e.g., ECCS, EDG.
- Failure to align FLSR (expected to dominate the FLSR unavailability) has the highest contribution to the CDF among the post initiator HFEs. This is due to the high failure probability (0.3) and crediting FLSR in risk significant scenarios, e.g., SBO, HCTL.
- Except the FLSR, failures of manual depressurisation have relatively high risk contributions among the post-initiator HFEs.
- Other post-initiator HFEs are not the important contributors. The reason is understood that the UK ABWR has sufficient automation (including passive actuation of COPS).
- Pre-initiator HFEs are not significant risk contributors to UK ABWR.

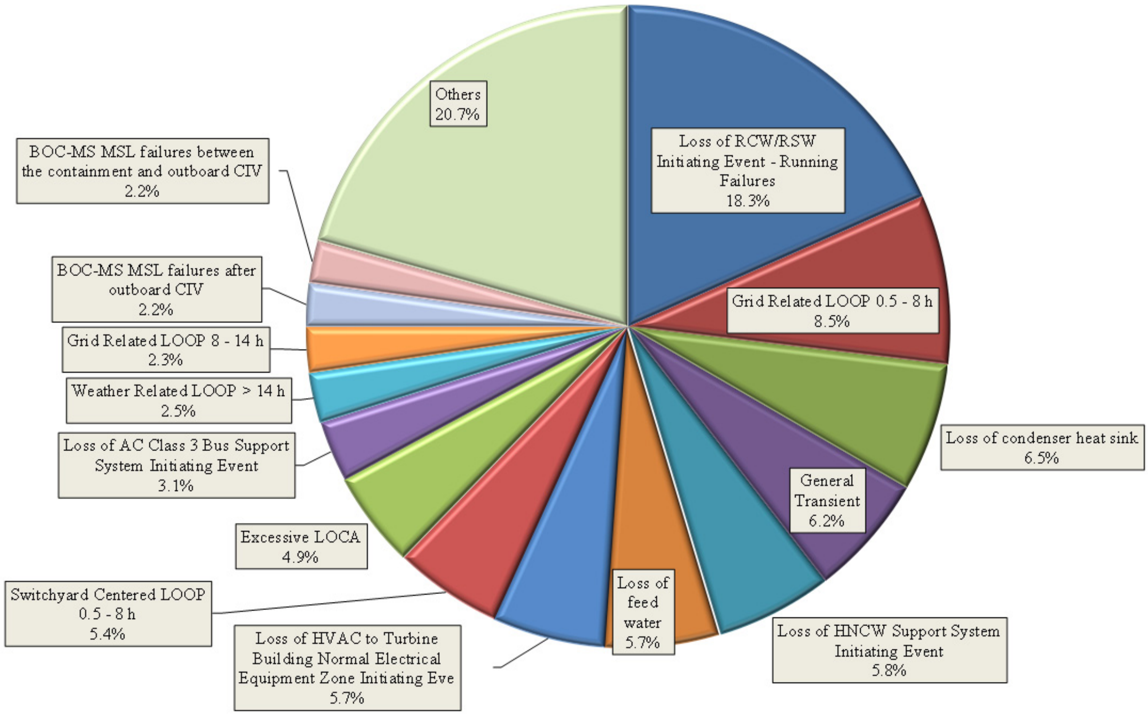

Figure 25.7.2-1 Contribution to IEAP CDF by Initiating Events

(IEs with less than 2 percent contribution are assigned to "others".)

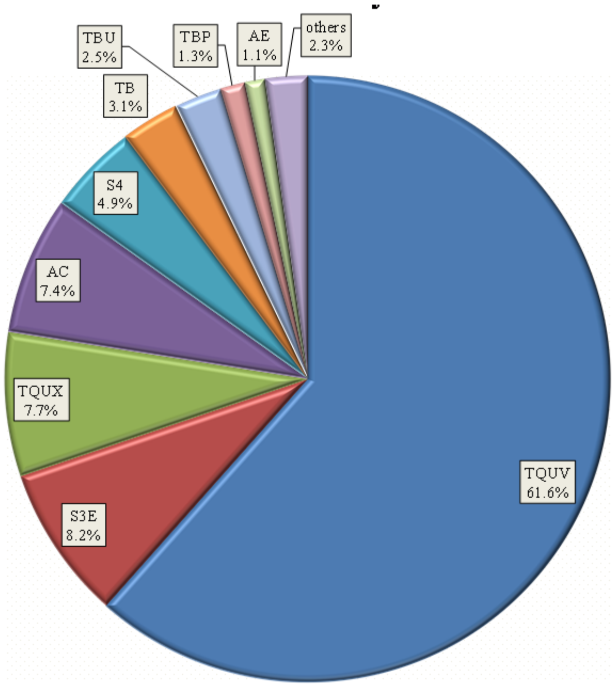

Figure 25.7.2-2 Contribution to IEAP CDF by Accident Classes

(Classes with less than 1 percent contribution are assigned to “others”.)

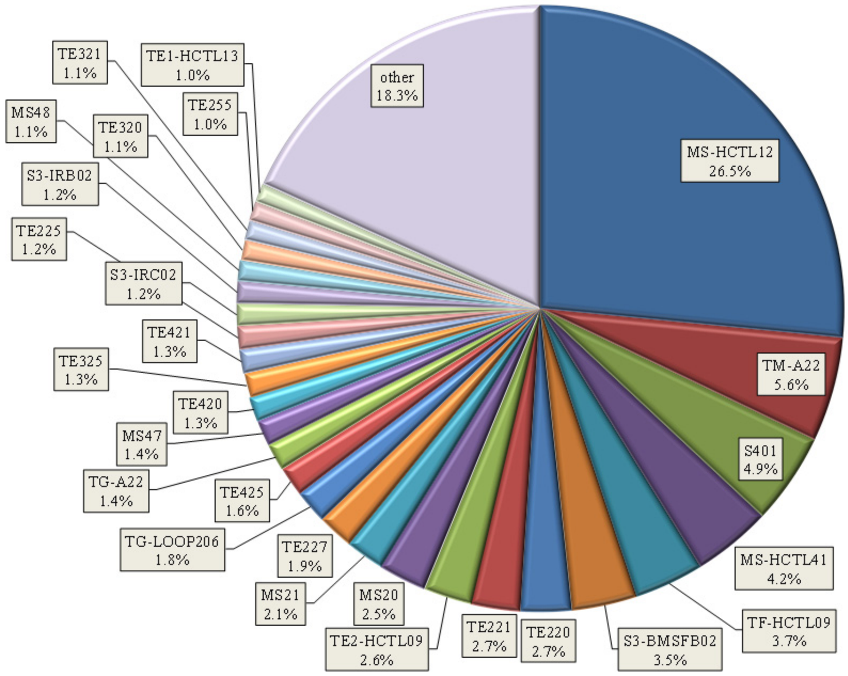

Figure 25.7.2-3 Contribution to IEAP CDF by Accident Sequences

(Sequences with less than 1 percent contribution are assigned to “others”.)

Table 25.7.2-1 Importance of Components (sorted by F-V) (1/3)

Rank	Event	F-V	RAW	Description
1	FL_PDS-TQUV	6.16E-01	1.00E+00	PDS tag for TQUV
2	PLANT-AVAILABILITY	4.79E-01	1.05E+00	Plant Availability Factor – At Power
3	FL_L1-MS-HCTL12	2.65E-01	1.00E+00	Sequence tag for L1-MS-HCTL12
4	FLSR-UNAVAILABLE	2.18E-01	1.51E+00	FLSR (Mobile Injection Facility) Unavailability
5	___-SLU-LF-__-HVAC	1.97E-01	1.97E+03	Software Failure induced CCF HVAC Loss of Function
6	IE-RCW_RSW	1.83E-01	8.17E-01	Loss of RCW/RSW Initiating Event - Running Failures
7	BBG-1	1.43E-01	4.60E+00	Loss of BBG-1
8	BBG-2	1.34E-01	4.37E+00	Loss of BBG-2
9	B21-SRV-CC-27-F001BEGKMSU	9.49E-02	1.11E+02	CCF of 2/7 Safety Relief Valve F001BEGKMSU Fail to Open
10	TEG2	8.53E-02	8.50E+00	Grid Related LOOP 0.5 to 8 h
11	FL_PDS-S3E	8.24E-02	1.00E+00	PDS tag for S3E
12	FL_PDS-TQUX	7.68E-02	1.00E+00	PDS tag for TQUX
13	FL_PDS-AC	7.37E-02	1.00E+00	PDS tag for AC
14	R43-DG_-RA-__-D/G-C	7.16E-02	3.76E+00	D/G-C Fail to Run > 1 Hour
15	CCF-BBG	7.15E-02	4.68E+01	CCF of loss of BBG
16	R43-DG_-RA-__-D/G-B	6.68E-02	3.57E+00	D/G-B Fail to Run > 1 Hour
17	___-SLU-LF-__-RPS	6.65E-02	6.66E+02	Software Failure induced CCF RPS Loss of Function
18	TM	6.48E-02	1.45E+00	Loss of condenser heat sink
19	E51-TPR-FS-__-C001	6.33E-02	2.29E+00	Turbine-Driven Pump (RCIC) C001 Fail to Start
20	IE-TG	6.16E-02	1.03E+00	General Transient

Table 25.7.2-1 Importance of Components (sorted by F-V) (2/3)

Rank	Event	F-V	RAW	Description
21	___-U41-BB-MT-B	6.03E-02	6.97E+00	Test and Maintenance Unavailability of HVAC-BB-B
22	___-E71-TL-MT-B_TRAIN	6.02E-02	6.96E+00	Test and Maintenance Unavailability of FLSS-B_TRAIN
23	SRV-C	5.92E-02	2.51E+00	Failure of SRV reclosure
24	R43-DG-RA-ALL_1_2_3	5.86E-02	4.12E+02	CCF of three components: R43-DG_-RA-__-D/G-A & R43-DG_-RA-__-D/G-B & R43-DG_-RA-
25	IE-HNCW	5.84E-02	9.42E-01	Loss of HNCW Support System Initiating Event
26	P41-SRR-PG-99-D001ABCDEFGHJ	5.80E-02	1.45E+06	CCF of 6/9 to 9/9 Strainer D001ABCDEFGHJ Plugged
27	P21-MPR-R_-66-C001ABCDEF	5.76E-02	1.45E+06	CCF of 6/6 Motor-Driven Pump (Running) C001ABCDEF Fail to Run
27	P41-MPR-R_-66-C001ABCDEF	5.76E-02	1.45E+06	CCF of 6/6 Motor-Driven Pump (Running) C001ABCDEF Fail to Run
29	TF	5.75E-02	1.87E+00	Loss of feed water
30	___-U41-BB-MT-A	5.74E-02	6.68E+00	Test and Maintenance Unavailability of HVAC-BB-A
31	___-E71-TL-MT-A_TRAIN	5.73E-02	6.68E+00	Test and Maintenance Unavailability of FLSS-A_TRAIN
32	IE-HVTE	5.71E-02	9.43E-01	Loss of HVAC to Turbine Building Normal Electrical Equipment Zone Initiating Eve
33	___-SLU-LF-__-RHR_RCW	5.61E-02	5.61E+02	Software Failure induced CCF RHR_RCW Loss of Function
34	FL_L1-TM-A22	5.58E-02	1.00E+00	Sequence tag for L1-TM-A22
35	TES2	5.42E-02	8.30E+00	Switchyard Centered LOOP 0.5 to 8 h
36	___-E71-SL-MT-FLSS	5.42E-02	5.43E+02	Test and Maintenance System Level Unavailability of FLSS
37	___-HFE-DF-TR	5.18E-02	1.97E+00	(All) HCTL-based Depressurisation and initiation of FLSS
38	FL_L1-S401	4.86E-02	1.00E+00	Sequence tag for L1-S401
38	FL_PDS-S4	4.86E-02	1.00E+00	PDS tag for S4
38	S4	4.86E-02	5.40E+06	Excessive LOCA

Table 25.7.2-1 Importance of Components (sorted by F-V) (3/3)

Rank	Event	F-V	RAW	Description
41	CP-TEMP	4.32E-02	2.40E+00	Conditional probability of outside air temperature to exceed acceptable limit
42	FL_L1-MS-HCTL41	4.20E-02	1.00E+00	Sequence tag for L1-MS-HCTL41
43	R43-DG_-RA-__-D/G-A	3.88E-02	2.50E+00	D/G-A Fail to Run > 1 Hour
44	FL_L1-TF-HCTL09	3.69E-02	1.00E+00	Sequence tag for L1-TF-HCTL09
45	FL_L1-S3-BMSFB02	3.46E-02	1.00E+00	Sequence tag for L1-S3-BMSFB02
46	C51-RS-NL-SRNM_ALL	3.32E-02	6.61E+00	CCF of all components in group 'C51-RS-NL-SRNM'
47	___-HFE-MD-TR	3.25E-02	1.07E+00	(Diagnosis/Intervention) Failure of manual RPV Depressurisation during transient
48	C-LOOP-2	3.24E-02	1.34E+01	Conditional LOOP 0.5 to 8 h
49	U41-TT-NL-TE130-140-150_1_2_3	3.15E-02	2.45E+02	CCF of three components: U41-TT_-NL-__-TE130 & U41-TT_-NL-__-TE140 & U41-TT_-NL-
50	FL_PDS-TB	3.11E-02	1.00E+00	PDS tag for TB
51	IE-AC-C3	3.07E-02	9.69E-01	Loss of AC Class 3 Bus Support System Initiating Event
52	___-E11-A_-MT	3.02E-02	3.99E+00	Test and Maintenance Unavailability of RHR-A
53	___-R44-A_-MT	2.98E-02	2.96E+00	Test and Maintenance Unavailability of BBG (A)
54	___-R44-B_-MT	2.84E-02	2.86E+00	Test and Maintenance Unavailability of BBG (B)
55	R22-BSA-LF-__-M/C-1A1	2.74E-02	8.23E+02	Bus (AC Power) M/C-1A1 Loss of Function
56	U41-FNR-R_-C804_1_2	2.74E-02	5.45E+03	CCF of two components: U41-FNR-R_-__-C804A & U41-FNR-R_-__-C804B
56	U41-FNR-R_-C805_1_2	2.74E-02	5.45E+03	CCF of two components: U41-FNR-R_-__-C805A & U41-FNR-R_-__-C805B
58	FL_L1-TE220	2.70E-02	1.00E+00	Sequence tag for L1-TE220
59	___-R43-C_-MT	2.70E-02	2.77E+00	Test and Maintenance Unavailability of DG-C
60	CST-INVENTORY	2.68E-02	1.44E+01	Failure probability for inadequate inventory in the CST

Table 25.7.2-2 Top Ten F-V Test and Maintenance Events

Rank	T&M Event	F-V	Description
1	___-E71-TL-MT-B_TRAIN	0.060	Test and Maintenance Unavailability of FLSS-B_TRAIN
1	___-U41-BB-MT-B	0.060	Test and Maintenance Unavailability of HVAC-BB-B
3	___-E71-TL-MT-A_TRAIN	0.057	Test and Maintenance Unavailability of FLSS-A_TRAIN
3	___-U41-BB-MT-A	0.057	Test and Maintenance Unavailability of HVAC-BB-A
5	___-E71-SL-MT-FLSS	0.054	Test and Maintenance System Level Unavailability of FLSS
6	___-E11-A_-MT	0.030	Test and Maintenance Unavailability of RHR-A
7	___-R44-A_-MT	0.030	Test and Maintenance Unavailability of BBG (A)
8	___-R44-B_-MT	0.028	Test and Maintenance Unavailability of BBG (B)
9	___-R43-C_-MT	0.027	Test and Maintenance Unavailability of DG-C
10	___-R43-B_-MT	0.025	Test and Maintenance Unavailability of DG-B

Table 25.7.2-3 Top Ten F-V Post-Initiator HFEs

Rank	HFE	F-V	Description
1	FLSR-UNAVAILABLE	0.22	FLSR (Mobile Injection Facility) Unavailability
2	___-HFE-DF-TR	0.052	(All) HCTL-based depressurisation and initiation of FLSS
3	___-HFE-MD-TR	0.033	(Diagnosis/Intervention) Failure of manual RPV depressurisation during transient
4	___-HFE-CV-FH	0.0035	(Diagnosis/Intervention) Failure of containment venting
5	___-HFE-CC-CG	0.0027	(Cognition) Cognition error for manual operation of core cooling
6	___-HFE-SR-AT	0.0018	(All) Manual opening SRVs and control water level during ATWS (NOTE: HEP set to 1.0)
7	___-HFE-HP-HB	0.0016	Failure of manual initiation and operation of HPCF-C with AM panel
8	___-HFE-PC-01	0.0013	(All) Failure of set of necessary operations for using PCS for FW-CS
9	___-HFE-FL-IN	0.0007	(Diagnosis/Intervention) Failure of depressurisation and FLSS initiation
10	___-HFE-LT-CG	0.0006	(Cognition) Cognition Error for Manual Operation of Long Term Heat Removal

Table 25.7.2-4 Top Ten F-V Pre-Initiator HFEs

Rank	HFE	F-V	Description
1	C73-HFE-FP-__-ARICBPDPBB13A	2.2E-03	Failure of DC Circuit Breaker 72-PDPBB13-ARI-A
1	C73-HFE-FP-__-ARICBPDPBB23B	2.2E-03	Failure of DC Circuit Breaker 72-PDPBB23-ARI-B
1	C73-HFE-FP-__-CBPDPBB13	2.2E-03	Failure of DC Circuit Breaker 72-BB1-PDPBB13
1	C73-HFE-FP-__-CBPDPBB23	2.2E-03	Failure of DC Circuit Breaker 72-BB2-PDPBB23
5	B21-HFE-MC-44-LTFLSSABCD	2.2E-03	CCF Miscalibration of Low Reactor Water Level Transmitter B21-LT-FLSS-ABCD
6	B21-HFE-MC-44-LT003JKLM	1.3E-03	CCF Miscalibration of Level Transmitter B21-LT003JKLM
7	B21-HFE-MC-44-PT009ABCD	9.1E-04	CCF Miscalibration of Pressure Transmitter B21-PT009ABCD
8	E71-HFE-CL-__-MVF005B	8.4E-04	Manual valve F005B left close
9	E71-HFE-CL-__-MVF005A	8.0.E-04	Manual valve F005A left close
10	B21-HFE-MC-44-PTFLSSABCD	3.8E-04	CCF Miscalibration of Low Reactor Pressure Transmitter B21-PT-FLSS-ABCD

Table 25.7.2-5 Top Ten RAW Pre-Initiator HFEs

Rank	HFE	RAW	Description
1	B21-HFE-MC-44-PT009ABCD	9.2E+01	CCF Miscalibration of Pressure Transmitter B21-PT009ABCD
2	B21-HFE-MC-44-LTFLSSABCD	4.4E+01	CCF Miscalibration of Low Reactor Water Level Transmitter B21-LT-FLSS-ABCD
3	B21-HFE-MC-44-PTFLSSABCD	3.9E+01	CCF Miscalibration of Low Reactor Pressure Transmitter B21-PT-FLSS-ABCD
4	B21-HFE-MC-44-LT003JKLM	2.7E+01	CCF Miscalibration of Level Transmitter B21-LT003JKLM
5	E71-HFE-CL-__-MVF005B	1.0E+01	Manual valve F005B left close
6	E71-HFE-CL-__-MVF005A	9.9E+00	Manual valve F005A left close
7	B21-HFE-MCH-44-PT007ABCD	9.0E+00	CCF Miscalibration of Pressure Transmitter B21-PT007ABCD (Higher drift)
8	C73-HFE-FP-__-ARICBPDPBB13A	3.2E+00	Failure of DC Circuit Breaker 72-PDPBB13-ARI-A
9	C73-HFE-FP-__-ARICBPDPBB23B	3.2E+00	Failure of DC Circuit Breaker 72-PDPBB23-ARI-B
10	C73-HFE-FP-__-CBPDPBB13	3.2E+00	Failure of DC Circuit Breaker 72-BB1-PDPBB13
11	C73-HFE-FP-__-CBPDPBB23	3.2E+00	Failure of DC Circuit Breaker 72-BB2-PDPBB23

Table 25.7.2-6 System Level Importance (sorted by F-V)

Rank (F-V)	Systems	FV	RAW
1	CONTROL PANELS(DIGITAL,RPS)	4.24E-01	2.23E+04
2	FLSS	3.24E-01	5.21E+02
3	BBG	2.83E-01	4.72E+01
4	DG	2.20E-01	4.01E+02
5	FLSR	2.18E-01	2.47E+02
6	SRV	1.83E-01	3.58E+02
7	HVAC	1.78E-01	9.90E+03
8	NB	1.37E-01	2.33E+05
9	RSW	1.29E-01	1.31E+00
10	RCIC	7.67E-02	2.26E+00
11	RCW	7.48E-02	1.98E+01
12	MC	5.30E-02	1.46E+04
13	HNCW	5.06E-02	5.43E+00
14	RHR	4.65E-02	4.00E+04
15	DC	1.90E-02	1.56E+04
16	PC	1.46E-02	7.81E+03
17	HECW	1.27E-02	2.54E+01
18	HPCF	9.10E-03	8.68E+02
19	ATWS	8.96E-03	4.44E+02
20	TCW	6.88E-03	3.74E+00
21	C,FDW/AO	5.79E-03	3.50E+00
22	SLC	5.74E-03	6.40E+00
23	IA	4.22E-03	4.61E+00
24	ANT	3.62E-03	1.85E+02
25	MUWC	3.60E-03	4.43E+00
26	MCC	3.10E-03	2.68E+03
27	TSW	1.87E-03	1.62E+00
28	VSS	1.54E-03	1.00E+00
29	EECW	1.53E-03	8.60E+00
30	RFC	1.26E-03	1.33E+00
31	CRD	1.11E-03	1.93E+00
32	FCVS	1.02E-03	1.11E+00
33	SPTM	4.19E-04	1.25E+00
34	AS	2.10E-04	1.10E+00
35	MS	1.64E-04	2.38E+00
36	TGS	1.06E-04	1.10E+00
37	LDS	2.35E-05	1.00E+00
38	SA	7.75E-06	1.01E+00
39	HPIN	6.45E-06	1.00E+00
40	OG	2.84E-06	1.04E+00
41	MUWP	1.53E-06	1.00E+00
42	CUW	1.07E-06	1.00E+00
43	AC	6.91E-07	1.00E+00
44	CW	7.67E-08	1.00E+00

Table 25.7.2-7 System Level Importance (sorted by RAW)

Rank (RAW)	Systems	FV	RAW
1	NB	1.37E-01	2.33E+05
2	RHR	4.65E-02	4.00E+04
3	CONTROL PANELS(DIGITAL,RPS)	4.24E-01	2.23E+04
4	DC	1.90E-02	1.56E+04
5	MC	5.30E-02	1.46E+04
6	HVAC	1.78E-01	9.90E+03
7	PC	1.46E-02	7.81E+03
8	MCC	3.10E-03	2.68E+03
9	HPCF	9.10E-03	8.68E+02
10	FLSS	3.24E-01	5.21E+02
11	ATWS	8.96E-03	4.44E+02
12	DG	2.20E-01	4.01E+02
13	SRV	1.83E-01	3.58E+02
14	FLSR	2.18E-01	2.47E+02
15	ANT	3.62E-03	1.85E+02
16	BBG	2.83E-01	4.72E+01
17	HECW	1.27E-02	2.54E+01
18	RCW	7.48E-02	1.98E+01
19	EECW	1.53E-03	8.60E+00
20	SLC	5.74E-03	6.40E+00
21	HNCW	5.06E-02	5.43E+00
22	IA	4.22E-03	4.61E+00
23	MUWC	3.60E-03	4.43E+00
24	TCW	6.88E-03	3.74E+00
25	C,FDW/AO	5.79E-03	3.50E+00
26	MS	1.64E-04	2.38E+00
27	RCIC	7.67E-02	2.26E+00
28	CRD	1.11E-03	1.93E+00
29	TSW	1.87E-03	1.62E+00
30	RFC	1.26E-03	1.33E+00
31	RSW	1.29E-01	1.31E+00
32	SPTM	4.19E-04	1.25E+00
33	FCVS	1.02E-03	1.11E+00
34	TGS	1.06E-04	1.10E+00
34	AS	2.10E-04	1.10E+00
36	OG	2.84E-06	1.04E+00
37	SA	7.75E-06	1.01E+00
38	MUWP	1.53E-06	1.00E+00
39	CUW	1.07E-06	1.00E+00
40	AC	6.91E-07	1.00E+00
41	HPIN	6.45E-06	1.00E+00
42	LDS	2.35E-05	1.00E+00
42	CW	7.67E-08	1.00E+00
42	VSS	1.54E-03	1.00E+00

25.7.3 Results of Level 2 PSA

The purpose of this section is to document the calculation of the Large Release Frequency (LRF) and Large Early Release Frequency (LERF) due to events that occur when the plant is operating at power. This section covers internal events at power only.

The calculation of the total LRF and LERF is performed as a single top gate. For analysis of contribution by release category and sequence, sequence markers are utilised.

Quantification of the model results in the following key outputs:

- Overall Large Release Frequency (LRF),
- Large release frequency as a function of:
  - Plant damage state,
  - Release category including containment leakage and containment venting,
  - Accident sequence,
- Importance characterisation of individual events (in terms of industry standard risk importance measures, for example: Fussell-Vesely; Risk Achievement Worth, and so forth) relative to the large release frequency,
- Overall Large Early Release Frequency (LERF), and
- Sensitivity and uncertainty analysis are summarised in Section 25.7.5.

25.7.3.1 Large Release Frequency

Large Release Frequency (LRF) is evaluated by the sum of the relevant release category frequencies as shown in Table 25.5.3-4.

The total LRF resulting from accident sequences for internal events at power is 5.20E-08 /y when a truncation value of 1.0E-17 /y is used for quantification [Ref-25.10].

Note that the total LRF is different from that in Table 25.7.3-2 due to the deletion of non-minimal cutsets.

Figure 25.7.3-1 shows the LRF contribution by PDS.

(1) Top 5 Contribution by PDS

1. AC (LOCAs with failure of reactivity control): 38 %
2. S3E (including BOC/ISLOCA): 32 %
3. S4 (RPV rupture): 19 %
4. TC (Transients with failure of reactivity control): 3 %
5. TB (Loss of Class 1 and Class 2 AC power resulting in high pressure core damage after 8 hours of RCIC operation): 3 %

In PDSs of AC (No. 1) and S4 (No. 3), containment fails prior to core damage. In S3E (No. 2), Fission Products is released at the same timing of core damage since containment isolation function fails. Since mitigation of accident progression is not expected for these PDSs, LRF of each PDS corresponds to CDF of each PDS.

(2) Release Category Contribution

An additional perspective on the Level 2 PSA results is provided by examining the relative contributions to the LRF of the release categories whose designator is shown in Table 25.5-20. Figure 25.7.3-2 illustrates the LRF contribution of each release category in the form of a pie chart. Note that containment leakage and containment venting are not the release categories with the large fission product release.

The top release categories each contributing 1 percent or more of the total LRF for internal events at power are discussed.

1. The largest release category contributor is C, which involves failure of reactivity control, resulting in large release of Fission Products in the short term. This release category contributes approximately 42 percent to LRF. This category includes the most significant accident sequence AC-22 in which lower drywell injection and upper drywell spray by FLSS succeeds after containment failure due to overpressure.
2. The second largest frequency contributing release category is BYPASS, which involves ISLOCAs, BOCs or the hydrogen combustion during de-inerted operation, resulting in large release of Fission Products in the short term. This release category contributes approximately 33 percent to LRF. In this release category, the hydrogen combustion during de-inerted operation is included. The contribution of the hydrogen combustion is approximately 3 percent to the frequency of this release category.
3. The third largest contributing release category is RR. This category involves RPV rupture, resulting in large release of Fission Products in the short term. This release category contributes approximately 21 percent to LRF.
4. The fourth largest contributing release category is LTSBO. This category involves loss of Class 1 and Class 2 AC power resulting in high pressure core damage after 8 hours of RCIC operation, resulting in large release of Fission Products in the late phase. This release category contributes approximately 17 percent to LRF.

(3) Accident Sequences

A total of 159 large release accident sequences are quantified when the truncation value is 1E-14. Among them, 14 sequences match the definition of significant accident sequences.

Figure 25.7.3-3 illustrates the LRF contribution for the top 14 accident sequences in the form of a bar chart. The significant accident sequences which contribute to LRF are described below.

1. L2-S3E-95 The PDS “S3” is ISLOCA or BOC with failure of RPV injection, resulting in low pressure core damage in short term with containment bypass. Manual containment isolation is not credited in the Level 2 PSA according to the insight of reviewing HFEs task. This accident sequence with release category “BYPASS” has a frequency of 1.38E-08 /y, and represents 29 percent of total LRF.

   - The most dominant location of pipe break is the RHR shutdown cooling suction line. It is assumed that the pipe break results in failure of the same division of ECCS due to the adverse effect of steam.

2. L2-AC-22 The PDS “AC” is a LOCA (A) followed by failure of reactivity control. In this PDS, water injection to the core is successful but both control rod insertion and boric acid injection into the core are failed. Heat removal from the PCV is insufficient because core power of this case is

kept relatively higher than that of decay heat. As a result, PCV overpressure failure occurs prior to core damage. In this sequence, debris on lower drywell is not able to be cooled sufficiently in spite of success of LDF. This accident sequence with release category “C” has a frequency of 1.32E-08 /y, and represents 28 percent of total LRF. The most dominant failure is CCF of software for RPS and pressure transmitter for ARI/RPT. The reason of high contribution by this sequence is only relying on high RPV pressure signal for ARI / RPT and the shared set of pressure instrumentation (Class 2) by ARI /RPT. Those failures results in the loss of reactivity control.

3. L2-S4-29 Excessive LOCA inside containment, directly resulting in low pressure core damage in short term and containment fails due to overpressure at the same timing of RPV rupture. After RPV breach, LDF is successful. This accident sequence with release category “RR” has a frequency of 8.52E-09 /y, and represents 18 percent of total LRF.
4. L2-AC-25 This sequence is similar to L2-AC-22. The difference is only the availability of PCV spray. Some of cutsets in two sequences are almost the same. As sequence marker is adopted in this Level 2 PSA, some duplicated cutsets provide the conservative LRF as shown in the below table. This is not limited in this sequence but causes the conservative results in other similar sequences. This accident sequence with release category “C” has a frequency of 2.00E-09 /y, and represents 4.2 percent of total LRF.

Top cutset of L2-AC-22	Top cutset of L2-AC-25
Loss of condenser heat sink	Loss of condenser heat sink
Software Failure induced CCF RPS Loss of Function	Software Failure induced CCF RPS Loss of Function
CCF of all components in group 'B21-PS-FM-PS609'	CCF of all components in group 'B21-PS-FM-PS609'
Containment Failure Mode tag for C	Containment Failure Mode tag for C
Sequence tag for L1-TM-A22	Sequence tag for L1-TM-A22
Sequence tag for L2-AC-22	Sequence tag for L2-AC-25
PDS tag for AC	PDS tag for AC
Release Category tag for C	Release Category tag for C
-	*Failure of upper drywell injection with FLSS*

Italic font in L2-AC-25 means the difference from L2-AC-22.

5. L2-S3E_LOOPA-95 This sequence is similar to L2-S3E-95. The difference is the occurrence of consequential LOOP. This accident sequence with release category “BYPASS” has a frequency of 1.49E-09 /y, and represents 3.1 percent of total LRF.
6. L2-TB_C1SBO-09 Since the UK ABWR has the RCIC as high pressure ECCS and the RCIC can inject water to the core without AC power, the core can be covered by water until DC power for the RCIC flow rate control is depleted. The timing of core damage is long term because of the water injection by RCIC. Since the depletion of DC power also disables manual depressurisation by opening SRVs as well as RCIC operation, the RPV pressure is high at the moment of core damage. In this sequence, PCV fails following RPV breach due to overpressure/overtemperature though debris on lower drywell is cooled sufficiently by LDF after PCV failure. This accident

sequence with release category "LTSBO" has a frequency of 1.39E-09 /y, and represents 2.9 percent of total LRF.

7. L2-AC-29 This sequence is similar to L2-AC-22. The difference is only success of systems for the lower drywell injection (L2-AC-22: FLSS, L2-AC-29: LDF). This accident sequence with release category "C" has a frequency of 1.19E-09 /y, and represents 2.5 percent of total LRF.
8. L2-TC-01 The PDS "TC" is a transient (T) followed by "PCV failure in short term and prior to core damage due failure of reactivity control (C). In this PDS, water injection to the core is successful but both control rod insertion and boric acid injection into the core are failed. Heat removal from the PCV is not enough because core power of this case is kept relatively higher than that of decay heat. As a result, PCV overpressure failure occurs prior to core damage. In this sequence, RPV is intact with the success of RPV depressurisation and FLSS. This accident sequence with release category "C" has a frequency of 7.66E-10 /y, and represents 1.6 percent of total LRF.
9. L2-AC-32 This sequence is similar to L2-AC-25. The difference is only success of systems for the lower drywell injection (L2-AC-25: FLSS, L2-AC-32: LDF). This accident sequence with release category "C" has a frequency of 6.59E-10 /y, and represents 1.4 percent of total LRF.
10. L2-AE-27 The PDS "AE" is defined as a LOCA inside containment (including consequential LOCAs) with failure of RPV injection, resulting in low pressure core damage in the short term. In this sequence, PCV fails before RPV breach due to overpressure/overtemperature though debris on lower drywell is cooled sufficiently by FLSS after PCV failure. This accident sequence with release category "RR" has a frequency of 5.33E-10 /y, and represents 1.1 percent of total LRF.
11. L2-AC-01 This sequence is similar to L2-TC-01. The difference is the PDS. The initiating event of AC is LOCA inside containment (including consequential LOCAs). This accident sequence with release category "C" has a frequency of 4.79E-10 /y, and represents 1.0 percent of total LRF.
12. L2-S4-30 Excessive LOCA inside containment, directly resulting in low pressure core damage in short term and containment fails due to overpressure at the same timing of RPV rupture. This accident sequence with release category "RR" has a frequency of 4.50E-10 /y, and represents 0.9 percent of total LRF.
13. L2-TC_LOOPA-01 This sequence is similar to L2-TC- 01. The difference is offsite power which is unavailable in this sequence. This accident sequence with release category "C" has a frequency of 3.77E-10 /y, and represents 0.8 percent of total LRF.
14. L2-AE_LOOPA-27 This sequence is similar to L2-AE-27. The difference is offsite power which is unavailable in this sequence. This accident sequence with release category "RR" has a frequency of 3.70E-10 /y, and represents 0.8 percent of total LRF.

(4) Importance analysis

This section provides the top events in various categories of events for F-V and, when insights can be realised, RAW, and discusses the basis for these events being significant [Ref-25.10].

Components important to safety (top ten F-V events)

The components important to safety are discussed here by F-V values. The initiating events, CCF events, test and maintenance events, human failure events and other marker events (e.g., IE markers, sequence markers, class markers, etc.) are excluded.

<u>Pressure Switch for ATWS signal</u>

The highest F-V events are Pressure switches for the RPV pressure instrumentation “B21-PS_-FM-__-PS609A”, “B21-PS_-FM-__-PS609B”, “B21-PS_-FM-__-PS609C” and “B21-PS_-FM-__-PS609D” which both ARI and RPT depend on. The F-V is 0.0337. ATWS signal is important given RPS fails in LOCA because these failures results in containment failure prior to core damage.

<u>FLSR</u>

The fifth highest F-V event is FLSR-UNAVAILABLE “FLSR (Mobile Injection Facility) Unavailability”: the F-V is 0.0318. The high F-V is due to the credit to FLSR in many significant sequences, e.g., HCTL condition, Loss of Class 1 and Class 2 AC power (station blackout), lower drywell flooding, upper drywell spray, and also the bounding probability of FLSR failure (including human error probability).

NOTE: This basic event probability represents the overall failure probability of FLSR including mechanical failure and human error because the detail design is not provided in GDA.

<u>Backup Building Generators</u>

The sixth highest F-V event is BBG-1 “Loss of BBG-1”: the F-V is 0.0281. The seventh highest F-V event is BBG-2 “Loss of BBG-2”: the F-V is 0.0280. These high F-V values are due to many significant sequences involving loss of Class 1 AC power, e.g., LOOP with failure of all EDGs.

The F-V of BBG-2 is slightly lower than that of BBG-1 despite of the same failure probability and symmetrical system structure of FLSS divisions 1 and 2 including support systems. The reason of this slight difference is because the hardened containment venting system is supported by BBG-1 while filtered containment venting system is supported by BBG-2. The importance of BBG-2 is slightly lower because the filtered containment venting system includes COPS which does not need AC power.

NOTE: The basic event probability is calculated by quantifying the EDG system fault tree including DGFO because the detail design is not provided in GDA.

<u>Output Logic Unit</u>

The eighth to tenth highest F-V events are output logic unit for MSIV closure. The F-V is 0.0145. These high F-V values are due to the failures related to containment isolation function. These failures disable containment isolation or BOC isolation, resulting in PCI or BYPASS release category.

Test and Maintenance events

Top ten F-V Test and Maintenance Events

Rank	T&M Event	F-V	Description
1	___-C73-AT-MT-DIVC	3.40E-02	Test and Maintenance Unavailability of ATWS System DIVC
2	___-C73-AT-MT-DIVD	3.38E-02	Test and Maintenance Unavailability of ATWS System DIVD
3	___-C73-AT-MT-DIVB	3.37E-02	Test and Maintenance Unavailability of ATWS System DIVB
4	___-C73-AT-MT-DIVA	3.35E-02	Test and Maintenance Unavailability of ATWS System DIVA
5	___-E22-C_-MT	1.42E-02	Test and Maintenance Unavailability of HPCF-C
6	___-E22-B_-MT	1.42E-02	Test and Maintenance Unavailability of HPCF-B
7	___-U41-BB-MT-B	1.36E-02	Test and Maintenance Unavailability of HVAC-BB-B
8	___-U41-BB-MT-A	1.35E-02	Test and Maintenance Unavailability of HVAC-BB-A
9	___-E71-TL-MT-B_TRAIN	1.35E-02	Test and Maintenance Unavailability of FLSS-B_TRAIN
10	___-E71-TL-MT-A_TRAIN	1.35E-02	Test and Maintenance Unavailability of FLSS-A_TRAIN

Rank 1 to 4:

T&M event of ATWS signal is important because ATWS related PDS (AC, S12C, TC, TC-HP, TC-LP and TCN) contributes 42 percent to the total LRF.

Rank 5:

T&M event of HPCF-C is important. This is because HPCF-C is only credited for the core cooling in ISLOCA at RHR-B shutdown mode suction line. S3E contributes 32 percent to the total LRF.

Rank 6:

T&M event of HPCF-B is important. This is because HPCF-B is only credited for the core cooling in ISLOCA at RHR-C shutdown mode suction line. S3E contributes 32 percent to the total LRF.

Rank 7:

T&M events of FLSS room cooling (HVAC-BB-B) have the identical probability because FLSS has some important functions of core injection, lower drywell injection and upper drywell spray.

The F-V of BBG-2 is slightly lower than that of BBG-1 despite of the same failure probability and symmetrical system structure of FLSS divisions 1 and 2 including support systems. The reason of this slight difference is because the hardened containment venting system is supported by BBG-1 while filtered containment venting system is supported by BBG-2. The importance of BBG-2 is slightly lower because the filtered containment venting system includes COPS which does not need AC power. Therefore the combination of T&M events of HVAC-BB-B and BBG-1 is more important than the combination of HVAC-BB-A and BBG-2.

Rank 8:

T&M events of FLSS room cooling (HVAC-BB-A) have the identical probability because FLSS has some important functions of core injection, lower drywell injection and upper drywell spray.

The F-V of BBG-2 is slightly lower than that of BBG-1. The reason is described in Rank 7.

Rank 9:

T&M event of FLSS-B is important. FLSS has some important functions of core injection, lower drywell injection and upper drywell spray. The reason of slightly lower F-V for FLSS-B than that for FLSS-A is the same as BBG.

Rank 10:

T&M event of FLSS-A is important. FLSS has some important functions of core injection, lower drywell injection and upper drywell spray. The reason of slightly lower F-V for FLSS-B than that for FLSS-A is the same as BBG.

Post-Initiator HFEs

Top ten F-V Post-Initiator HFEs

Rank	HFE	F-V	Description
1	___-HFE-FU-L2	5.55E-02	Failure of upper drywell injection with FLSS
2	___-HFE-FL-L2	3.55E-02	Failure of lower drywell injection with FLSS
3	FLSR-UNAVAILABLE	3.18E-02	FLSR (Mobile Injection Facility) Unavailability
4	DAG	1.43E-02	Loss of DAG
5	___-HFE-DC-L2	1.39E-02	(Cognition) Failure of cognition for debris cooling and containment heat removal
6	___-HFE-CV-FH	1.94E-03	(Diagnosis/Intervention) Failure of containment venting
7	___-HFE-FS-L2	9.40E-04	Failure of depressurisation and FLSS initiation for Level 2 PSA
8	___-HFE-LT-CG	8.60E-04	(Cognition) Cognition Error for Manual Operation of Long Term Heat Removal
9	___-HFE-DF-TR	4.20E-04	(All) HCTL-based depressurisation and initiation of FLSS
10	___-HFE-MD-TR	3.00E-04	(Diagnosis/Intervention) Failure of manual RPV depressurisation during transient

Rank 1:

Failure of manual upper drywell spray with FLSS has high F-V because many of the OP/OT sequences include this HFE. When PCV spray is successful, release rate of particle Fission Products would be low. This event has the highest F-V value among all the post-initiator HFEs.

Rank 2:

Failure of manual lower drywell injection with FLSS has high F-V because many of the OP/OT and CCI sequences include this HFE.

Rank 3:

FLSR-UNAVAILABLE includes the mechanical failure and human failure. Since FLSR-UNAVAILABLE is considered to be dominated by human error probability, this event is included in this section. Loss of FLSR has high F-V as described in (a).

Rank 4:

Since DAG is considered to be dominated by human error probability, this event is included in this section. DAG is additionally credited for RHR AC power in Level 2 PSA.

Rank 5:

Failure of cognition for debris cooling has high F-V because many of the OP/OT and CCI sequences include this HFE.

All other post-initiator HFEs do not match the definition of significant basic event in terms of F-V (above 0.005).

Pre-Initiator HFEs

Top ten F-V Pre-Initiator HFEs

Rank	HFE	F-V	Description
1	C73-HFE-FP-__-ARICBPDPBB13A	7.69E-03	Failure of DC power supply Circuit Breaker 72-PDPBB13-ARI-A
2	C73-HFE-FP-__-ARICBPDPBB23B	7.69E-03	Failure of DC power supply Circuit Breaker 72-PDPBB23-ARI-B
3	C73-HFE-FP-__-CBPDPBB13	7.69E-03	Failure of DC power supply Circuit Breaker 72-BB1-PDPBB13
4	C73-HFE-FP-__-CBPDPBB23	7.69E-03	Failure of DC power supply Circuit Breaker 72-BB2-PDPBB23
5	B21-HFE-MC-44-LT003JKLM	6.59E-03	CCF Miscalibration of Level Transmitter B21-LT003JKLM
6	B21-HFE-MC-44-PT009ABCD	4.67E-03	CCF Miscalibration of Pressure Transmitter B21-PT009ABCD
7	B21-HFE-MC-44-LTFLSSABCD	4.36E-03	CCF Miscalibration of Low Reactor Water Level Transmitter B21-LT-FLSS-ABCD
8	B21-HFE-MC-__-LT003L	1.29E-03	Miscalibration of Level Transmitter B21-LT003L
9	B21-HFE-MC-__-LT003M	1.28E-03	Miscalibration of Level Transmitter B21-LT003M
10	B21-HFE-MC-__-LT003K	1.27E-03	Miscalibration of Level Transmitter B21-FT003K

Rank 1 to 4:

Pre-initiator HFEs of DC power for FLSS has the identical probability. FLSS has some important functions of core injection, lower drywell injection and upper drywell spray.

Rank 5:

CCF Miscalibration of Level Transmitter B21-LT-003 has the fifth highest RAW. The failure of RPV level instrumentation causes loss of both ARI and RPT. When transient events occur, Reactor Coolant Pressure Boundary (RCPB) pressure would exceed 120 percent of its design pressure due to failure to trip all 10 RIPs, resulting in consequential LOCA (AC) as well as core damage and containment failure.

All other post-initiator HFEs do not match the definition of significant basic event .

Phenomena

Top ten F-V Phenomena

Rank	Phenomena	F-V	Description
1	H2_COMBUSTION	6.04E-03	Probability of Hydrogen Combustion
2	MCCI-CONT_DB	4.82E-03	Containment failure by MCCI without water filling prior to RPV breach
3	FCI-EX-WET	3.16E-03	Probability of ex-vessel FCI with pedestal water filling
4	MCCI-PART_DB	3.06E-03	Containment failure by MCCI with water filling prior to RPV breach
5	DCH	2.50E-04	Containment failure by DCH
6	FCI-EX-DRY	4.00E-05	Probability of ex-vessel FCI without pedestal water filling
7	FCI-IN-VES	2.00E-05	Probability of in-vessel FCI

Rank 1:

The second highest F-V event is the hydrogen combustion. It is important because all core damage events during de-inerted operation lead to containment failure due to hydrogen combustion.

All other post-initiator HFEs do not match the definition of significant basic event.

Release category

Top ten F-V Release Category

Rank	Phenomena	F-V	RAW	Description
1	FL_RC-C	4.18E-01	1.00E+00	Release Category tag for C
2	FL_RC-BYPASS	3.31E-01	1.00E+00	Release Category tag for BYPASS
3	FL_RC-RR	2.14E-01	1.00E+00	Release Category tag for RR
4	FL_RC-LTSBO	3.09E-02	1.00E+00	Release Category tag for Long term SBO
5	FL_RC-OP/OT3	3.83E-03	1.00E+00	Release Category tag for OP/OT3
6	FL_RC-SPBYP	1.50E-03	1.00E+00	Release Category tag for SPBYP
7	FL_RC-PCI2	5.60E-04	1.00E+00	Release Category tag for PCI2
8	FL_RC-OP/OT2	4.50E-04	1.00E+00	Release Category tag for OP/OT2
9	FL_RC-DCH	2.50E-04	1.00E+00	Release Category tag for DCH
10	FL_RC-PE2	1.70E-04	1.00E+00	Release Category tag for PE2
11	FL_RC-CCI2	8.00E-05	1.00E+00	Release Category tag for CCI2
12	FL_RC-RE2	0.00E+00	1.00E+00	Release Category tag for RE2

System Level Importance

All the basic events that have the unique system ID (including CCF events and pre-initiator HFEs) are assigned to one of the "systems" specified by the system ID. Some miscellaneous basic events, e.g., BBG-1, BBG-2, CCF-BBG, are manually checked for specifying the relevant system, e.g., BBG.

The system level F-V value of a system is calculated by summing up the F-V values of all the relevant basic events. The system level RAW of a system is represented by the highest RAW among those of the relevant basic events.

System-level importance was evaluated. To develop system level importance, the F-V was assessed by summing all F- V of the basic events in the system.

The observations are as below.

F-V:

- CONTROL PANEL (digital C&I components and software) has the highest F-V. The reasons for the high F-V are wide use of digital C&I system for Class 1 and Class 3 safety systems and the conservative modelling of software CCF in terms of probability and consequence.
- Nuclear Boiler system (NB) has the second highest F-V. Various SSCs are involved: reactor vessel instrumentation, MSIVs, FW-A line check valves (forms injection line of RHR-A, FLSS, FLSR), etc.
- ATWS signal has the third highest F-V. This is why AC has the highest frequency (38 percent).
- FLSS has the fourth highest F-V. This is because FLSS is credited for most of the PDSs and because most of the TQUV (62 percent of total CDF) sequences involves failure of FLSS. Also, FLSS is credited to prevent not only degraded core injection but also debris cooling and containment spray.
- RHR has the fifth highest F-V. Loss of RHR is credited to prevent not only degraded core injection but also debris cooling and containment spray. Also, injection line to RPV is commonly used for FLSS and FLSR.
- BBG has the sixth highest F-V. The reason for the high F-V is that BBG is needed to operate FLSS and manual containment venting in the LOOP conditions (including conditional LOOP and LOOP caused by support system initiators) which occurs in high frequency.
- HPCF has the seventh highest F-V. Rank 4 of top ten F-V CCF events are the MOV CCF at RHR shutdown mode suction lines. In this initiating event, the different division of HPCF is only credited for the core cooling. S3E contributes 32 percent to the total LRF.
- HVAC has the eighth highest F-V. The reason for the high F-V is that room cooling is needed to wide range of systems/components, e.g., HPCF, RHR, FLSS, EDG, BBG, RCW, RSW, TCW, TSW, FW, CS, PCS, buses, controllers.
- EDG has the ninth highest F-V. The reason for the high F-V is that EDGs is needed to operate HPCF, RHR and their support systems in the LOOP conditions (including conditional LOOP and LOOP caused by support system initiators) which occurs in high frequency.
- FLSR has the tenth highest F-V. FLSR has high unavailability (0.3) FLSR is credited to prevent not only degraded core injection but also debris cooling and containment spray.

RAW:

- CRD has the highest RAW. This is due to the small failure probability to insert CRs.
- RCW and RSW are ranked 2 and 3, respectively. This is obvious because loss of all RCW/RSW disables most of the frontline systems except reactivity control systems, RCIC (for 8 hours), FLSS, FLSR, Containment venting system, ADS, RDCF.
- CONTROL PANEL (digital C&I components and software) has the fourth highest RAW. The reasons for the high F-V are wide use of digital C&I system for Class 1 and Class 3 safety systems and the

conservative modelling of software CCF in terms of probability and consequence.

- Nuclear Boiler system (NB) has the fifth highest RAW. Various SSCs are involved: reactor vessel instrumentation, MSIVs, FW-A line check valves (forms injection line of RHR-A, FLSS, FLSR), etc.
- RHR has the sixth highest RAW. Loss of RHR is credited to prevent not only degraded core injection but also debris cooling and containment spray. Also, injection line to RPV is commonly used for FLSS and FLSR.
- SRV has the seventh highest RAW. Various failure modes of SRVs are involved: failure to open for depressurisation, failure to open for pressure boundary protection, failure to reclose causing SORV.
- DC has the eighth highest RAW. Class 1 DC and Class 2 DC are divided independently. However, if both DC powers are not available, the credit of almost mitigation systems except RCIC, LDF, COPS etc. cannot be taken.
- VSS has the ninth highest RAW. VSS failure results in the containment failure due to overpressure in the short term.
- FLSS has the tenth highest RAW. This is because FLSS is credited for most of the PDSs sequences. Also, FLSS is credited to prevent not only degraded core injection but also debris cooling and containment spray.

25.7.3.2 Frequency of Each Release Category

Table 25.7.3-1 shows the frequency of each release category, which is used for Level 3 PSA as shown in Section 25.6.

The results of Level 2 PSA consist of radioactive release frequencies and source term. The results of source term analysis are shown in Section 25.5.3.5.

The top accident sequences for each release category are described below.

Note: The release category frequencies are evaluated from the cutsets with the truncation value of 1E-14. Therefore, the frequencies in Table 25.7.3-1 are different from those in Table 25.7.3-2.

1. Filtered Containment Venting (FVV and FVP_I, II and III) The top accident sequence for this release category is L2-TQUV-02. TQUV denotes "Transient (T) including planned manual shutdown and special initiators", "followed by failures of feedwater system (Q)", "failures of high pressure ECCS (U)", and "failures of low pressure ECCS (V)". Though RPV fails, followed by core damage, LDF and COPS are successful in Level 2 PSA, resulting in release category FVP. This accident sequence has a frequency of 4.17E-08 /y. This release category does not correspond to the large release [Ref-25.10].

2. Containment Leakage (KV and KP_I, II and III) The top accident sequence for this release category is L2-TQUV-01_C1SBO-001. TQUV denotes "Transient (T) including planned manual shutdown and special initiators", "followed by failures of feedwater system (Q)", "failures of high pressure ECCS (U)", and "failures of low pressure ECCS (V)" and C1SBO denotes loss of Class1 AC power. Low pressure injection after core damage is successful and then RPV is intact. In addition, long term heat removal is successful, resulting in containment leakage (KV). This accident sequence has a frequency of 4.96E-09 /y. This release category does not correspond to the large release.

3. Early Containment Failure (ALL_IV_except_BYPASS) The top accident sequence for this release category is L2-AC-22. The PDS "AC" is LOCA (A) followed by failure of reactivity control. In this PDS, water injection to the core is successful but both control rod insertion and boric acid injection into the core are failed. Heat removal from the PCV is not enough because core power of this case is kept relatively higher than that of decay heat. As a result, PCV overpressure failure occurs prior to core damage. In this sequence, debris on lower drywell is not able to be cooled sufficiently in spite of success of LDF. This accident sequence with release category "C" has a frequency of 1.32E-08 /y. This release category corresponds to the large early release.

4. Containment Bypass (ALL VII and BYPASS_except III) The top accident sequence for this release category is L2-S3E-95. The PDS "S3" is ISLOCA or BOC with failure of RPV injection, resulting in low pressure core damage in short term with containment bypass. Manual containment isolation is not credited in the Level 2 PSA according to the insight of reviewing HFEs task. This accident sequence with release category "BYPASS" has a frequency of 1.38E-08 /y. This release category corresponds to the large early release. In this category, hydrogen combustion during de-inerted operation is included. The contribution of the hydrogen combustion is approximately 3 percent to the frequency of this release category.

5. RPV Rupture (ALL_IX_except BYPASS and RR and RR_LD_AE) The top accident sequence for this release category is L2-S4-29. Excessive LOCA inside containment, directly resulting in low pressure core damage in short term and then containment fails due to overpressure at the same timing of RPV rupture. After RPV breach, LDF is successful. This accident sequence with release category "RR" has

a frequency of 8.52E-09 /y. This release category corresponds to the large early release.

6. Late Containment Failure (OP/OT_I) The top accident sequence for this release category is L2-TQUV-74. The PDS "TQUV" is transient including planned manual shutdown and special initiators, followed by failures of high pressure and low pressure injection systems. The success of RPV depressurisation results in low pressure in short term. In this sequence, failure of water injection to degraded core results in RPV breach and debris on lower drywell is not able to be cooled sufficiently though FLSS lower drywell injection is successful. This accident sequence with release category "OP/OT1" has a frequency of 2.14E-09 /y. This release category does not correspond to the large release.

7. Late Containment Failure (OP/OT_PS_I, II and III) The top accident sequence for this release category is L2-TQUV-69. The PDS "TQUV" is transient including planned manual shutdown and special initiators, followed by failures of high pressure and low pressure injection systems. The success of RPV depressurisation results in low pressure in short term. In this sequence, failure of water injection to degraded core results in RPV breach and debris on lower drywell is not able to be cooled sufficiently though LDF and PCV spray are successful. This accident sequence with release category "OP/OT_PS" has a frequency of 9.57E-10 /y. This release category does not correspond to the large release.

8. Direct Debris Interaction (DDI_II) The top accident sequence for this release category is L2-TQUX-102. TQUX denotes "Transient (T) including planned manual shutdown and special initiators", "followed by failures of feedwater system (Q)", "failures of high pressure ECCS (U)", and "failures of reactor depressurisation (X)". RPV depressurisation after core damage fails, resulting in RPV breach. Containment failure due to Direct Debris Interaction occurs followed by RPV high pressure breach. This accident sequence with release category "DDI" has a frequency of 9.96E-10 /y. This release category does not correspond to the large early release.

9. Late Containment Failure (ALL_VI_except_BYPASS) The top accident sequence for this release category is L2-TW-LP_LOOPA-27. The PDS "TW-LP" is transient including planned manual shutdown and special initiators, followed by failure of heat removal measures from the PCV. It means that water injection to the core at low pressure is successful and PCV heat removal fails. The core can be covered by water for a long period. However, decay heat generated from the core cannot be removed from the PCV, and the PCV is finally failed by overpressure due to decay heat accumulation. Failures of equipment installed in Reactor Building are assumed due to the adverse effect of steam at a timing of the PCV failure. As a result, PCV fails due to overpressure at first and then, it results in core damage at RPV low pressure due to loss of water injection to the core. In this sequence, debris on lower drywell is cooled sufficiently by LDF. This accident sequence with release category "OP/OT4" has a frequency of 1.00E-09 /y. This release category does not correspond to the large release.

10. Long term SBO (Containment Failure w/o Spray III) The top accident sequence for this release category is L2-TB_C1SBO-09. Since UK ABWR has the RCIC as high pressure ECCS and the RCIC can inject water to the core without AC power, core can be covered by water till DC power for the RCIC flow rate control is depleted. The timing of core damage is long term because of the water injection of the RCIC. Since the depletion of DC power also disables manual depressurisation by opening SRVs as well as RCIC operation, the RPV pressure is high at the moment of core damage. In this sequence, PCV fails following RPV breach due to overpressure/overtemperature though debris on lower drywell is cooled sufficiently by LDF after PCV failure. This accident sequence with release

category "LTSBO" has a frequency of 1.39E-09 /y. This release category corresponds to the large release.

11. Molten Core Concrete Interaction (CCI_I) The top accident sequence for this release category is L2-TQUV-90. The PDS "TQUV" is Transient including planned manual shutdown and special initiators, followed by failures of high pressure and low pressure injection systems. The success of RPV depressurisation results in low pressure in short term. In this sequence, failure of water injection to degraded core results in RPV breach and debris on lower drywell is not able to be cooled sufficiently in spite of success of LDF. This accident sequence with release category "CCI1" has a frequency of 1.00E-09 /y. This release category does not correspond to the large release.

12. Containment Venting (VV and VP_I, II and III) The top accident sequence for this release category is L2-TQUV-03. TQUV denotes "Transient (T) including planned manual shutdown and special initiators", "followed by failures of feedwater system (Q)", "failures of high pressure ECCS (U)", and "failures of low pressure ECCS (V)". Low pressure injection after core damage is successful, resulting in RPV intact. In addition, PCV spray and hardened containment venting is successful, resulting in VP. This accident sequence has a frequency of 3.27E-10 /y. This release category does not correspond to the large release.

13. Late Containment Failure (ALL_V_except_BYPASS) The top accident sequence for this release category is L2-AW-LP-01. The PDS "AW-LP" is LOCAs, followed by failure of heat removal measures from the PCV. It means that water injection to the core at low pressure is successful and PCV heat removal fails. The core can be covered by water for a long period. However, decay heat generated from the core cannot be removed from the PCV, and the PCV is finally failed by overpressure due to decay heat accumulation. Failures of equipment installed in Reactor Building are assumed due to the adverse effect of steam at a timing of the PCV failure. As a result, PCV fails due to overpressure at first and then, it results in core damage at RPV low pressure due to loss of water injection to the core. In this sequence, FLSS injection to the damaged core is successful. This accident sequence with release category "OP/OT3" has a frequency of 1.00E-09 /y. This release category corresponds to the large release.

14. Ex-vessel Fuel-Coolant Interaction (PE_I) The top accident sequence for this release category is L2-TQUV-63. TQUV denotes "Transient (T) including planned manual shutdown and special initiators", "followed by failures of feedwater system (Q)", "failures of high pressure ECCS (U)", and "failures of low pressure ECCS (V)". Low pressure injection after core damage fails, resulting in RPV breach. Although lower drywell injection by FLSS is successful, containment fails due to FCI. This accident sequence with release category "PE1" has a frequency of 1.29E-10 /y. This release category does not correspond to the large release.

15. S/P Bypass (ALL_VIII_except_BYPASS) The top accident sequence for this release category is L2-TNQUV-01. TNQUV denotes a "non-LOCA event with SRV tailpipe break in the W/W airspace," followed by failures of failures of high pressure injection, and failures of low pressure injection, resulting in low pressure core damage in short term with suppression pool bypass. PCV failure occurs when the RPV is depressurised due to the suppression pool bypass. This accident sequence with release category "SPBYP" has a frequency of 6.67E-11 /y. This release category corresponds to the large release.

16. PCV Isolation failure (PCI_I) The top accident sequence for this release category is L2-TQUV-92.

PCV is not isolated due to the failure of PCIS at the timing of core damage, resulting in PCV isolation failure (PCI1). This accident sequence has a frequency of 4.86E-11 /y. This release category does not correspond to the large release.

17. PCV Isolation Failure (PCI_II) The top accident sequence for this release category is L2-TQUX-132. PCV is not isolated due to the failure of PCIS at the timing of core damage, resulting in PCV isolation failure (PCI2). This accident sequence has a frequency of 9.89E-12 /y. This release category corresponds to the large release.

18. Late Containment Failure (OP/OT_II) The top accident sequence for this release category is L2-TBU-11. TBU denotes “Loss of offsite power transient (T)”, “failure to start all EDGs (B)”, and “failure to start RCIC (U)”. RPV depressurisation is successful after core damage, however RPV injection fails due to no Class 1 and Class 2 AC power. After that, debris fall into lower drywell. Although LDF is successful, debris cooling is not sufficiently, resulting “OP/OT2”. This accident sequence has a frequency of 6.46E-12 /y. This release category corresponds to the large release.

19. Direct Containment Heating The top accident sequence for this release category is L2-TQUX-101. TQUX denotes “Transient (T) including planned manual shutdown and special initiators”, “followed by failures of feedwater system (Q)”, “failures of high pressure ECCS (U)”, and “failures of reactor depressurisation (X)”. RPV depressurisation after core damage fails, resulting in RPV breach. Containment failure due to DCH occurs followed by RPV high pressure breach. This accident sequence with release category “DCH” has a frequency of 7.21E-12 /y. This release category corresponds to the large release.

20. Ex-vessel Fuel-Coolant Interaction (PE_II) The top accident sequence for this release category is L2-AE-33. AE denotes a Loss of Coolant Accident with failure of RPV injection, resulting in low pressure core damage in short term. Low pressure injection after core damage fails, resulting in RPV breach. Although lower drywell injection by FLSS is successful, containment fails due to FCI. This accident sequence with release category “PE2” has a frequency of 4.32E-12 /y. This release category corresponds to the large release.

21. Molten Core Concrete Interaction (CCI_II) The top accident sequence for this release category is L2-AE-43. AE denotes a Loss of Coolant Accident with failure of RPV injection, resulting in low pressure core damage in short term. Low pressure injection after core damage fails, resulting in RPV breach. The failures of Lower drywell injection by FLSS, LDF and PCV spray by FLSS lead to containment failure due to MCCI. This accident sequence with release category “CCI2” has a frequency of 1.17E-12 /y. This release category corresponds to the large release.

22. In-vessel Fuel-Coolant Interaction (RE_I) The top accident sequence for this release category is L2-TQUV-91. TQUV denotes “Transient (T) including planned manual shutdown and special initiators”, “followed by failures of feedwater system (Q)”, “failures of high pressure ECCS (U)”, and “failures of low pressure ECCS (V)”. After core damage, failure of low pressure injection causes the core support plate failure, resulting in in-vessel FCI. This accident sequence with release category “RE1” has a frequency of 9.85E-13 /y. This release category does not correspond to the large release.

23. In-vessel Fuel-Coolant Interaction (RE_II) The top accident sequence for this release category is L2-AE-44. AE denotes a Loss of Coolant Accident with failure of RPV injection, resulting in low pressure core damage in short term. After core damage, failure of low pressure injection causes the core support

plate failure, resulting in in-vessel FCI. This accident sequence with release category “RE2” has a frequency of 2.98E-14 /y. This release category corresponds to the large release.

25.7.3.3 Large Early Release Frequency

The release categories which correspond to Large Early Release are shown below [Ref-25.10].

- Late Containment Failure (OP/OT_II)
- In-vessel Fuel-Coolant Interaction (RE_II)
- Ex-vessel Fuel-Coolant Interaction (PE_II)
- Direct Containment Heating (DCH_II)
- PCV Isolation Failure (PCI_II)
- Molten Core Concrete Interaction (CCI_II)
- RPV Rupture (ALL_IX_except BYPASS and RR and RR_LD_AE)
- Containment Bypass (ALL VII and BYPASS_except III)
- S/P Bypass (ALL_VIII_except_BYPASS)
- Early Containment Failure (ALL_IV_except_BYPASS)

Some duplicated cutsets are included due to inclusion of some non-minimal cutsets such as sequence marker, release category marker, etc. (marker events). That provides the conservative frequency of a release category. In order to exclude the conservativeness, the cutsets for each release category is developed and then all marker events in the cutsets are deleted. The frequencies of release categories without marker events are provided in Table 25.7.3-2. LERF is evaluated by the sum of the frequencies for the relevant release categories.

Total LERF for UK ABWR is 4.43E-08 /y when a truncation value of 1.0E-16 /y or 1.0E-17 /y is used for quantification. Containment Leakage (KV and KP_I, II and III), Containment Venting (VV and VP_I, II and III) and Filtered Containment Venting (FVV and FVP_I, II and III) are evaluated with the truncation value of 1.0E-16 /y because these release categories have large number of cutsets. The remaining release categories are evaluated with the truncation value of 1.0E-17 /y. The discussion on above release categories are described in Section 25.7.3.2.

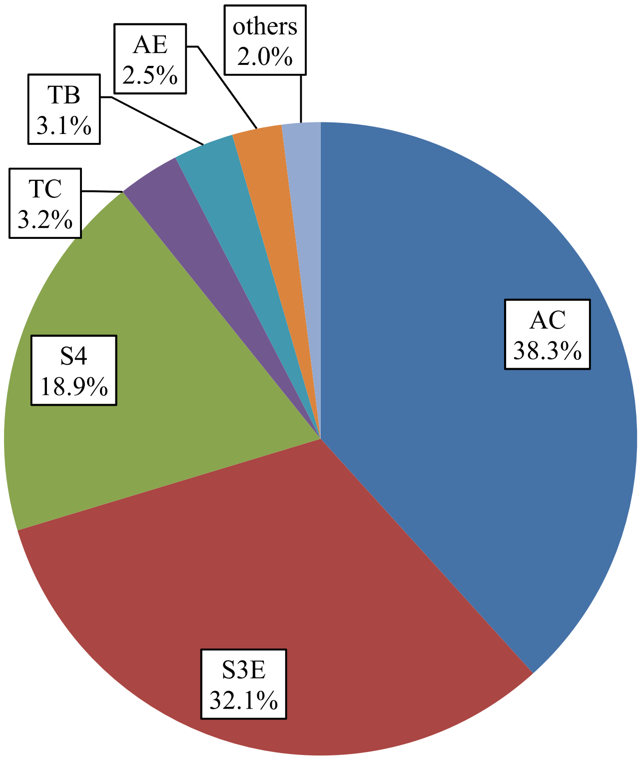

Figure 25.7.3-1 LRF Contribution by Plant Damage State

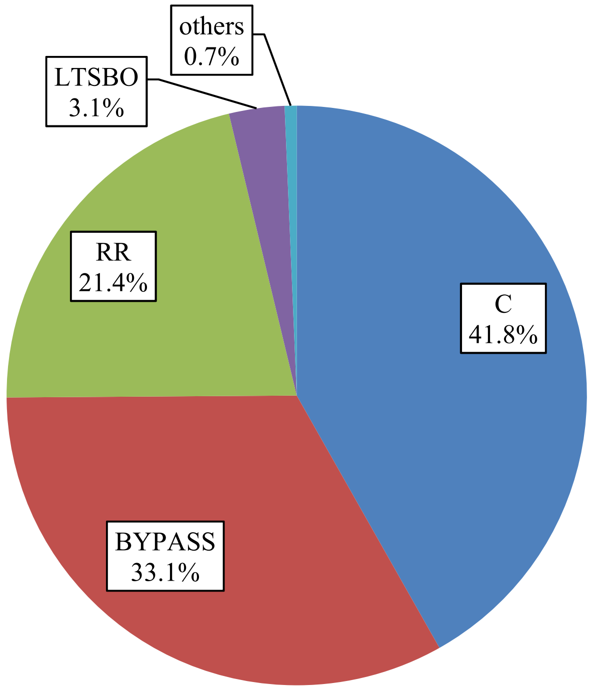

Figure 25.7.3-2 LRF Contribution by Release Category

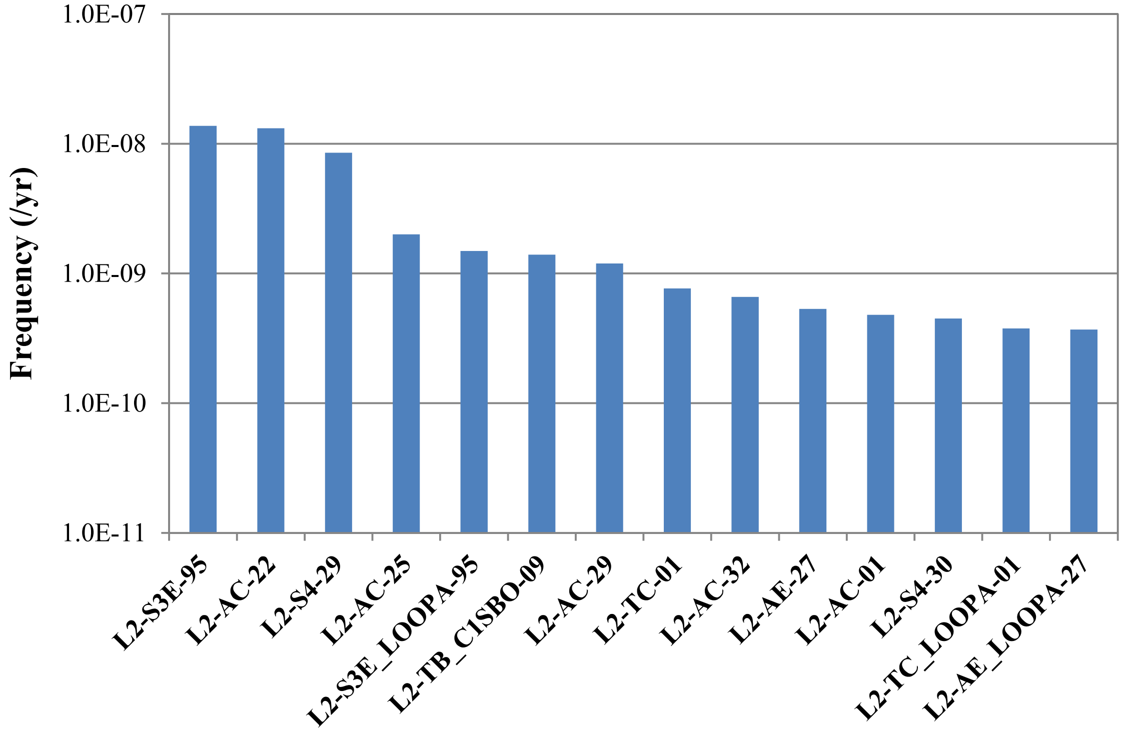

Figure 25.7.3-3 LRF by Significant Accident Sequences

Table 25.7.3-1 Radioactive Release Frequencies and Dominant Sequences Frequencies (1/4)

Rank	Release Category	Frequency (/y)	Rank for sequence	Sequence No. in Level 2	Frequency (/y)
1	Filtered Containment Venting (FVV and FVP_I, II and III) (Release category No. 3 in Table 25.7.3-2)	1.26E-07	1	L2-TQUV-02	4.17E-08
			2	L2-TQUV-70	2.25E-08
			3	L2-TQUV-53	1.67E-08
			4	L2-TQUV-65	1.05E-08
			5	L2-TQUX-002	5.03E-09
2	Containment Leakage (KV and KP_I, II and III) (Release category No. 1 in Table 25.7.3-2)	3.18E-08	1	L2-TQUX_C1SBO-001	4.96E-09
			2	L2-TQUV_C1SBO-64	3.92E-09
			3	L2-TQUV-01	3.68E-09
			4	L2-TBU-01	3.11E-09
			5	L2-TQUX-001	2.59E-09
3	Early Containment Failure (ALL_IV_except_BYPASS) (Release category No. 4 in Table 25.7.3-2)	1.99E-08	1	L2-AC-22	1.32E-08
			2	L2-AC-25	2.00E-09
			3	L2-AC-29	1.19E-09
			4	L2-TC-01	7.66E-10
			5	L2-AC-32	6.59E-10
4	Containment Bypass (ALL VII and BYPASS_except III) (Release category No. 13 in Table 25.7.3-2)	1.57E-08	1	L2-S3E-95	1.38E-08
			2	L2-S3E_LOOPA-95	1.49E-09
			3	L2-AN-01	2.15E-10
			4	L2-TQUV-93	1.58E-10
			5	L2-AC-41	3.77E-11
5	RPV Rupture (ALL_IX_except BYPASS and RR and RR_LD_AE) (Release category No. 12 in Table 25.7.3-2)	1.02E-08	1	L2-S4-29	8.52E-09
			2	L2-AE-27	5.33E-10
			3	L2-S4-30	4.50E-10
			4	L2-AE_LOOPA-27	3.70E-10
			5	L2-AE-30	7.86E-11
6	Late Containment Failure (OP/OT_I) (Release category No. 5-1 in Table 25.7.3-2)	2.82E-09	1	L2-TQUV-74	2.14E-09
			2	L2-TQUV_LOOPA-74	3.74E-10
			3	L2-TQUV_C1SBO-74	1.31E-10
			4	L2-TQUV-72	1.22E-10
			5	L2-TQUV_LOOPA-72	2.79E-11

Table 25.7.3-1 Radioactive Release Frequencies and Dominant Sequences Frequencies (2/4)

Rank	Release Category	Frequency (/y)	Rank for sequence	Sequence No. in Level 2	Frequency (/y)
7	Late Containment Failure (OP/OT_PS_I, II and III) (Release category No. 6 in Table 25.7.3-2)	2.43E-09	1	L2-TQUV-69	9.57E-10
			2	L2-TQUV_C1SBO-69	3.29E-10
			3	L2-TBU-06	2.96E-10
			4	L2-TQUV_LOOPA-69	2.06E-10
			5	L2-TBP-06	1.87E-10
8	Direct Debris Interaction (DDI_II) (Release category No. 15 in Table 25.7.3-2)	2.29E-09	1	L2-TQUX-102	9.96E-10
			2	L2-TBU-40	7.27E-10
			3	L2-TBU_C1SBO-35	2.09E-10
			4	L2-TQUX_C1SBO-102	1.41E-10
			5	L2-TQUX-114	1.27E-10
9	Late Containment Failure (ALL_VI_except_BYPASS) (Release category No. 5-3 in Table 25.7.3-2)	1.76E-09	1	L2-TW-LP_LOOPA-27	1.00E-09
			2	L2-TW-LP-01	3.33E-10
			3	L2-TW-LP-27	1.59E-10
			4	L2-TW-LP-17	1.01E-10
			5	L2-TW-LP_LOOPA-29	5.63E-11
10	Long term SBO (Containment Failure w/o Spray _III) (Release category No. 16 in Table 25.7.3-2)	1.47E-09	1	L2-TB_C1SBO-09	1.39E-09
			2	L2-TB_C1SBO-11	5.89E-11
			3	L2-TB_C1SBO-22	5.26E-12
			4	L2-TB-68	4.38E-12
			5	L2-TB_C1SBO-21	1.53E-12
11	Molten Core Concrete Interaction (CCI_I) (Release category No. 11-1 in Table 25.7.3-2)	1.22E-09	1	L2-TQUV-90	1.00E-09
			2	L2-TQUV_LOOPA-90	1.60E-10
			3	L2-TQUV_C1SBO-90	5.42E-11
			4	L2-TBP-27	6.68E-13
			5	L2-TBP_C1SBO-23	2.35E-13
12	Containment Venting (VV and VP_I, II and III) (Release category No. 2 in Table 25.7.3-2)	7.76E-10	1	L2-TQUV-03	3.27E-10
			2	L2-TQUV-54	1.46E-10
			3	L2-TQUV-71	1.18E-10
			4	L2-TQUX-003	5.63E-11
			5	L2-TQUV-66	4.66E-11

Table 25.7.3-1 Radioactive Release Frequencies and Dominant Sequences Frequencies (3/4)

Rank	Release Category	Frequency (/y)	Rank for sequence	Sequence No. in Level 2	Frequency (/y)
13	Late Containment Failure (ALL_V_except_BYPASS) (Release category No. 5-3 in Table 25.7.3-2)	1.82E-10	1	L2-AW-LP-01	1.09E-10
			2	L2-TW_LOOPA-27	3.81E-11
			3	L2-AW-LP-02	1.38E-11
			4	L2-TW_LOOPA-01	6.90E-12
			5	L2-AW-LP_LOOPA-27	3.75E-12
14	Ex-vessel Fuel-Coolant Interaction (PE_I) (Release category No. 8-1 in Table 25.7.3-2)	1.49E-10	1	L2-TQUV-63	1.29E-10
			2	L2-TQUV-75	1.47E-11
			3	L2-TQUV_C1SBO-63	1.92E-12
			4	L2-TQUV_LOOPA-63	1.07E-12
			5	L2-TQUV_C1SBO-75	1.05E-12
15	S/P Bypass (ALL_VIII_except_BYPASS) (Release category No. 14 in Table 25.7.3-2)	7.12E-11	1	L2-TNQUV-01	6.67E-11
			2	L2-TCN-01	1.62E-12
			3	L2-TCN_LOOPA-03	1.38E-12
			4	L2-TNQUX_LOOPA-01	4.01E-13
			5	L2-TCN_LOOPA-01	3.72E-13
16	PCV Isolation Failure (PCI_I) (Release category No. 10-1 in Table 25.7.3-2)	4.97E-11	1	L2-TQUV-92	4.86E-11
			2	L2-TBP-29	6.01E-13
			3	L2-TQUV_LOOPA-92	3.74E-13
			4	L2-TQUV_C1SBO-92	1.25E-13
17	PCV Isolation Failure (PCI_II) (Release category No. 10-2 in Table 25.7.3-2)	2.64E-11	1	L2-TQUX-132	9.89E-12
			2	L2-TC-HP_LOOPA-132	6.76E-12
			3	L2-TQUX_C1SBO-132	3.43E-12
			4	L2-TC-HP-132	2.96E-12
			5	L2-AE-45	2.46E-12
18	Late Containment Failure (OP/OT_II) (Release category No. 5-2 in Table 25.7.3-2)	2.12E-11	1	L2-TBU-11	6.46E-12
			2	L2-TQUX-008	3.76E-12
			3	L2-AE_LOOPA-39	2.76E-12
			4	L2-TBU_C1SBO-09	2.70E-12
			5	L2-AE-39	2.59E-12

Table 25.7.3-1 Radioactive Release Frequencies and Dominant Sequences Frequencies (4/4)

Rank	Release Category	Frequency (/y)	Rank for sequence	Sequence No. in Level 2	Frequency (/y)
19	Direct Containment Heating (DCH_II) (Release category No. 9 in Table 25.7.3-2)	1.21E-11	1	L2-TQUX-101	7.21E-12
			2	L2-TBU-39	2.13E-12
			3	L2-TQUX_C1SBO-101	9.81E-13
			4	L2-TBU_C1SBO-34	8.37E-13
			5	L2-TQUX-113	7.29E-13
20	Ex-vessel Fuel-Coolant Interaction (PE_II) (Release category No. 8-2 in Table 25.7.3-2)	8.23E-12	1	L2-AE-33	4.32E-12
			2	L2-AE_LOOPA-33	2.36E-12
			3	L2-TBU-12	8.21E-13
			4	L2-AE-40	4.67E-13
			5	L2-AE_LOOPA-40	1.32E-13
21	Molten Core Concrete Interaction (CCI_II) (Release category No. 11-2 in Table 25.7.3-2)	4.03E-12	1	L2-AE-43	1.17E-12
			2	L2-AE_LOOPA-43	1.07E-12
			3	L2-TBU-27	1.06E-12
			4	L2-TBU_C1SBO-23	7.00E-13
			5	L2-TQUX_LOOPA-090	2.68E-14
22	In-vessel Fuel-Coolant Interaction (RE_I) (Release category No. 7-1 in Table 25.7.3-2)	9.85E-13	1	L2-TQUV-91	9.85E-13
23	In-vessel Fuel-Coolant Interaction (RE_II) (Release category No. 7-2 in Table 25.7.3-2)	5.25E-14	1	L2-AE-44	2.98E-14
			2	L2-TBU-28	2.27E-14

Table 25.7.3-2 Radioactive Release Frequencies for Each Release Category

No.	Release Category	Frequency (/y)	LRF (/y)	LERF (/y)
1	Containment Leakage (KV and KP_I, II and III)	3.84E-08	-	-
2	Containment Venting (VV and VP_I, II and III)	1.05E-09	-	-
3	Filtered Containment Venting (FVV and FVP_I, II and III)	1.31E-07	-	-
4	Early Containment Failure (ALL_IV_except_BYPASS)	1.54E-08	X	X
5-1	Late Containment Failure (OP/OT_I)	4.39E-09	-	-
5-2	Late Containment Failure (OP/OT_II)	6.33E-11	X	X
5-3	Late Containment Failure (ALL_V_except_BYPASS)	3.47E-10	X	-
5-4	Late Containment Failure (ALL_VI_except_BYPASS)	3.93E-09	-	-
6	Late Containment Failure (OP/OT_PS_I, II and III)	4.21E-09	-	-
7-1	In-vessel Fuel-Coolant Interaction (RE_I)	5.41E-12	-	-
7-2	In-vessel Fuel-Coolant Interaction (RE_II)	4.22E-13	X	X
8-1	Ex-vessel Fuel-Coolant Interaction (PE_I)	2.51E-10	-	-
8-2	Ex-vessel Fuel-Coolant Interaction (PE_II)	1.50E-11	X	X
9	Direct Containment Heating (DCH_II)	2.41E-11	X	X
10-1	PCV Isolation Failure (PCI_I)	2.28E-10	-	-
10-2	PCV Isolation Failure (PCI_II)	8.06E-11	X	X
11-1	Molten Core Concrete Interaction (CCI_I)	1.91E-09	-	-
11-2	Molten Core Concrete Interaction (CCI_II)	1.77E-11	X	X
12	RPV Rupture (ALL_IX_except BYPASS and RR and RR_LD_AE)	1.00E-08	X	X
13	Containment Bypass (ALL VII and BYPASS_except III)	1.85E-08	X	X
14	S/P Bypass (ALL_VIII_except_BYPASS)	1.75E-10	X	X
15	Direct Debris Interaction (DDI_II)	2.70E-09	-	-
16	Long term SBO (Containment Failure w/o Spray _III)	1.58E-09	X	-
-	Total	2.34E-07	4.63E-08*	4.43E-08

X: Applicable

* The LRF is smaller than that shown in Section 25.7.3.1 due to the deletion of non-minimal cutsets.

25.7.4 Results of Level 3 PSA

The following results are the contributions of Internal Events at Power to probabilistic targets (NEEDP targets [Ref-25.58] equivalent to SAP Targets7, 8, and 9).

25.7.4.1 Facility Dose Bands

Table 25.7.4-1 presents the assessment against Target 8 for the IEAP (Modified Cases), based on the release category frequencies listed in Section 25.6. The allocation to facility dose band is made on the basis of the highest calculated dose across the range of distances considered to represent the closest locations with continuous occupancy by members of the public. The summated frequency for each Target 8 dose band is presented in Table 25.7.4-1 to enable comparison with the BSO and BSL for Target 8.

It can be seen that there is a margin to the BSO of an order of magnitude or more for all dose bands:

- There is no contribution to the 0.0001 to 0.001 Sv (0.1 to 1 mSv) and 0.01 to 0.1 Sv (10 to 100 mSv) facility dose bands.
- There is a negligible contribution to the 0.001 to 0.01 Sv (1 to 10 mSv) facility dose band.
- Release category 3 contributes 1.31E-07 /y to the 0.1 to 1 Sv (100 to 1,000 mSv) facility dose band. This is equivalent to 1.3 percent of the BSO.
- The remaining release categories contribute 6.49E-08 /y to the >1 Sv (>1,000 mSv) facility dose band. This is equivalent to 6.5 percent of the BSO.
  - Release category 2 falls in the > 1 Sv facility dose band (at 5.7 Sv) due to the addition of volatile iodine forms to the source term (elemental and organic forms).
  - All other release categories assigned to this facility dose band result in long term doses > 10 Sv.
  - Release categories contributing 4.39E-08 /y or 67.7 percent of the total (and individually > 1 percent of the BSO) for this dose band are:
    - Release category 13, containment bypass arising from ISLOCA / BOC, at 1.85E-08 /y (28.5 percent of the band frequency),
    - Release category 4, early containment failure arising from failure of reactivity control, at 1.54E-08 /y (23.7 percent of the band frequency),
    - Release category 12, early containment overpressurisation resulting from RPV rupture, at 1.00E-08 /y (15.4 percent of the band frequency).
    - It is considered that there may be some conservatism in the derivation of frequency for these three release categories.
  - A further seven release categories contribute an additional 1.98E-08 /y or 30.5 percent of the total (each at >1 percent of the total band frequency but < 1 percent of the BSO for this dose band):
    - Release category 5-1, late containment failure (TQUV), at 4.39E-09 /y (6.8 percent of the total),
    - Release category 6, late containment failure with PCV spray (AE), at 4.21E-09 /y (6.5 percent of the total),
    - Release category 5-4, late containment failure (TW-LP), at 3.93E-09 /y (6.1 percent of the total),
    - Release category 15, direct debris interaction (TQUX), at 2.70E-09 /y (4.2 percent of the total),
    - Release category 11-1, molten core concrete interaction (TQUV), at 1.91E-09 /y (2.9 percent of the total),

- o Release category 16, long term SBO (TB in-vessel FCI), at 1.58E-09 /y (2.4 percent of the total),
- o Release category 2, unfiltered containment venting (TQUV no DW sprays) at 1.05E-09 /y (1.6 percent of the total).

Table 25.7.4-1 Assessment Against Facility Dose Bands (Target 8) for IE at Power Leading to Fuel Melt

Facility Dose Band (Sv)	Release categories assigned to each dose band for IE at Power leading to fuel melt	Band Frequency (/y)	Percentage of Band BSO	BSO (/y)	BSL (/y)
0.0001 to 0.001				1.0E-2	1
0.001 to 0.01	1	3.84E-08	< 0.01 %	1.0E-3	1.0E-1
0.01 to 0.1				1.0E-4	1.0E-2
0.1 to 1	3	1.31E-07	1.3 %	1.0E-5	1.0E-3
> 1	2 (note 1) 4, 5-1, 5-2, 5-3, 5-4, 6, 7-1, 7-2, 8-1, 8-2, 9, 10-1, 10-2, 11-1, 11-2, 12, 13, 14, 15, 16 (note 2)	6.49E-08	6.5 %	1.0E-6	1.0E-4
Summated frequency of release categories /y		**2.34E-07**			

(note 1) – for this case the peak summated long term dose is < 10 Sv.

(note 2) – for all these cases the peak summated long term dose is > 10 Sv.

25.7.4.2 Individual Risk

The individual risk for each release category for IE at Power (Modified Cases) is calculated as the product of the conditional individual risk given in the PSA Summary Report [Ref-25.1] and the release category frequency given in Section 25.6. The conditional individual risk is calculated as the sum of the mean conditional risk of early fatal health effects and the mean conditional risk of long term (stochastic) health effects. The contribution of each release category to the overall risk from IE at Power is given in Table 25.7.4-2 and represented graphically in Figure 25.7.4-1.

The summated individual risk at 1 km from IE at Power is 5.94E-09 /y (i.e. about 0.6 percent of the BSO).

- It can be seen that three release categories contribute 5.05E-09 /y or 85 percent of the individual risk:
  - Release category 4, early containment failure arising from failure of reactivity control, is the largest contributor at 1.71E-09 /y (28.8 percent of the total),
  - Release category 13, containment bypass arising from ISLOCA / BOC contributes 2.28E-09 /y (38.4 percent of the total), and
  - Release category 12, early containment overpressurisation resulting from RPV rupture (with LDF success) contributes 1.05E-09 /y (17.8 percent of the total).

- A further five release categories contribute > 1 percent of the total, summated at 7.04E-10 /y or 11.8 percent of the total:
  - Release category 5-4, late containment failure (TW-LP), at 1.73E-10 /y (2.9 percent of the total),
  - Release category 16, long term SBO (TB in-vessel FCI), at 1.71E-10 /y (2.9 percent of the total),
  - Release category 5-1, late containment failure (TQUV), at 1.33E-10 /y (2.3 percent of the total),
  - Release category 6, late containment failure with PCV spray (AE), at 1.17E-10 /y (2.0 percent of the total), and
  - Release category 15, direct debris interaction (TQUX), at 1.08E-10 /y (1.8 percent of the total).

It is noted that the summated individual risk values reported here assume no offsite protective actions beyond restriction of foodstuffs, in line with ONR guidance. For most release categories it is likely that a number of short term and long term protective actions would be considered in the event of a release to reduce individual risk close to the site.

Table 25.7.4-2 Individual Risk of Fatality Close to the Site (Target 7) for IE at Power Leading to Fuel Melt

Release Category	Release Category Frequency (/y)	Individual risk of fatal health effects (/y)			Contribution to Total Risk at 1 km
		400 m	1,000 m	1,500 m	
1	3.84E-08	2.24E-13	8.79E-14	5.45E-14	0.00 %
2	1.05E-09	1.48E-11	4.28E-12	2.38E-12	0.07 %
3	1.31E-07	9.68E-11	3.43E-11	2.24E-11	0.58 %
4	1.54E-08	2.70E-09	1.71E-09	1.37E-09	28.8 %
5-1	4.39E-09	3.23E-10	1.33E-10	8.78E-11	2.25 %
5-2	6.33E-11	9.50E-12	4.94E-12	3.39E-12	0.08 %
5-3	3.47E-10	7.63E-11	5.03E-11	3.99E-11	0.85 %
5-4	3.93E-09	4.24E-10	1.73E-10	1.12E-10	2.92 %
6	4.21E-09	3.25E-10	1.18E-10	7.54E-11	1.98 %
7-1	5.41E-12	9.14E-13	5.04E-13	3.58E-13	0.01 %
7-2	4.22E-13	8.06E-14	5.12E-14	3.92E-14	0.00 %
8-1	2.51E-10	1.22E-11	3.95E-12	2.40E-12	0.07 %
8-2	1.50E-11	2.31E-12	1.26E-12	8.82E-13	0.02 %
9	2.41E-11	3.01E-12	1.31E-12	8.94E-13	0.02 %
10-1	2.28E-10	1.50E-11	5.26E-12	3.28E-12	0.09 %
10-2	8.06E-11	1.39E-11	9.09E-12	7.12E-12	0.15 %
11-1	1.91E-09	1.55E-10	5.71E-11	3.55E-11	0.96 %
11-2	1.77E-11	3.12E-12	1.59E-12	1.10E-12	0.03 %
12	1.00E-08	1.76E-09	1.05E-09	7.75E-10	17.8 %
13	1.85E-08	3.09E-09	2.28E-09	1.96E-09	38.4 %
14	1.75E-10	2.71E-11	1.65E-11	1.21E-11	0.28 %
15	2.70E-09	2.63E-10	1.08E-10	6.99E-11	1.82 %
16	1.58E-09	2.67E-10	1.71E-10	1.32E-10	2.88 %
Total individual risk (/y):		**9.58E-09**	**5.94E-09**	**4.71E-09**	
Total as % of BSO		**0.96 %**	**0.59 %**	**0.47 %**	
BSO		1.00E-06			
BSL		1.00E-04			

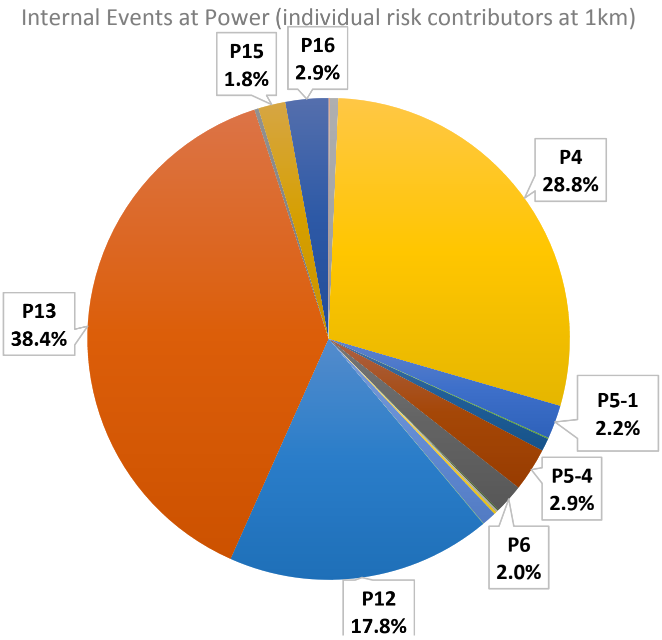

Figure 25.7.4-1 Contribution of Release Categories to the Individual Risk at 1 km for IE at Power Leading to Fuel Melt

25.7.4.3 Societal Risk

Table 25.7.4-3 presents the assessment against Target 9 for the IE at Power. This is based on the sum of the mean number of short term fatal health effects and the mean number of notional late fatalities in the UK population, as presented in the PSA Summary Report [Ref-25.1]. Where this is <100, a qualitative consideration of the potential onsite consequences is used to determine whether the threshold is exceeded. In this direct comparison approach, where the consequences threshold is exceeded the total release category frequency is added to the frequency of exceeding Target 9. Overall, for events leading to fuel melt, this direct comparison approach is deemed to give a conservative frequency for comparison against the BSL/BSO for Target 9.

The summated societal risk from IE at Power is 6.49E-08 /y (i.e. about 65 percent of the BSO).

All release categories except 1, and 3 are assigned as above the Target 9 threshold and the release categories contributing to this frequency are shown in Figure 25.7.4-2.

As all release categories assigned to the top facility dose band currently lead to an expectation of >100 deaths, with minimal protective actions, the breakdown is the same as that given in sub-section 25.7.2.1. It can be seen that:

- Three release categories contribute 4. 39E-08 /y (67.6 percent of the total):
  - Release category 13, containment bypass arising from ISLOCA / BOC contributes 1.85E-08 /y (28.5 percent of the total),
  - Release category 4, early containment failure arising from failure of reactivity control, is the largest contributor at 1.54E-08 /y (23.7 percent of the total), and
  - Release category 12, early containment overpressurisation resulting from RPV rupture contributes 1.00E-08 /y (15.4 percent of the total).
- Each of further seven release categories contributes >1 percent of the BSO:
  - Release category 5-1, late containment failure (TQUV), at 4.39E-09 /y (6.8 percent of the total),

- Release category 6, late containment failure with PCV spray (AE), at 4.21E-09 /y (6.5 percent of the total),
- Release category 5-4, late containment failure (TW-LP), at 3.93E-09 /y (6.1 percent of the total),
- Release category 15, direct debris interaction (TQUX), at 2.70E-09 /y (4.2 percent of the total),
- Release category 11-1, molten core concrete interaction (TQUV), at 1.91E-09 /y (2.9 percent of the total),
- Release category 16, long term SBO (TB in-vessel FCI), at 1.58E-09 /y (2.4 percent of the total), and
- Release category 2, unfiltered **c**ontainment venting (TQUV no DW sprays) at 1.05E-09 /y (1.6 percent of the total).

Table 25.7.4-3 Frequency of Exceeding the Societal Threshold (Target 9) for IE at Power Leading to Fuel Melt with Minimal Offsite Protective Actions

Release Category	Release category frequency (/y)	Frequency above Target 9 threshold (/y)	Contribution to Total Frequency
1	3.84E-08	0.00E+00	0.00 %
2	1.05E-09	1.05E-09	1.62 %
3	1.31E-07	0.00E+00	0.00 %
4	1.54E-08	1.54E-08	23.7 %
5-1	4.39E-09	4.39E-09	6.77 %
5-2	6.33E-11	6.33E-11	0.10 %
5-3	3.47E-10	3.47E-10	0.53 %
5-4	3.93E-09	3.93E-09	6.06 %
6	4.21E-09	4.21E-09	6.49 %
7-1	5.41E-12	5.41E-12	0.01 %
7-2	4.22E-13	4.22E-13	0.00 %
8-1	2.51E-10	2.51E-10	0.39 %
8-2	1.50E-11	1.50E-11	0.02 %
9	2.41E-11	2.41E-11	0.04 %
10-1	2.28E-10	2.28E-10	0.35 %
10-2	8.06E-11	8.06E-11	0.12 %
11-1	1.91E-09	1.91E-09	2.94 %
11-2	1.77E-11	1.77E-11	0.03 %
12	1.00E-08	1.00E-08	15.4 %
13	1.85E-08	1.85E-08	28.5 %
14	1.75E-10	1.75E-10	0.27 %
15	2.70E-09	2.70E-09	4.16 %
16	1.58E-09	1.58E-09	2.44 %
	Total Frequency: /y	**6.49E-08**	
	Total as % of BSO	**64.9 %**	
	BSO	**1.00E-07**	
	BSL	**1.00E-05**	

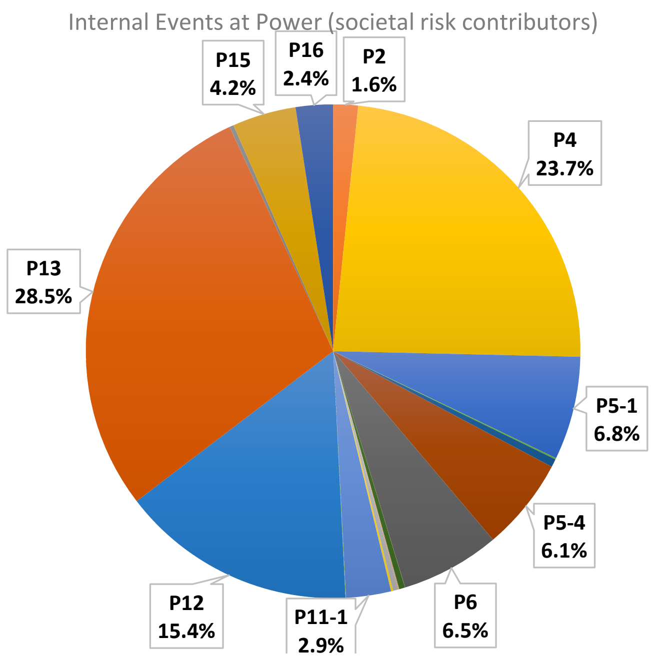

Figure 25.7.4-2 Contribution of Release Categories to the Frequency of Exceeding the Societal Risk Criterion for IE at Power Leading to Fuel Melt

25.7.5 Uncertainty and Sensitivity Analysis

25.7.5.1 Uncertainty Analysis

Uncertainty analyses were performed to support the robustness of the GDA PSA and the design of the UK ABWR.

Uncertainty analysis has been performed using the Monte Carlo sampling method which generated a probability density function and a cumulative probability function for the Core Damage Frequency (CDF) using the UNCERT function. The uncertainty distribution and error factors were captured in the Type Code (TC) table. In particular, the alpha-factor method was used to quantify CCFs in the UK ABWR PSA and is preferred as it enables a more robust modelling of uncertainty in the CCF parameters.

The mean CDF generated based on the sample size of 100,000 is 2.73E-07 /y.

Graphical results, as well as the uncertainty values (mean, 5 percent, 50 percent, 95 percent) for the uncertainty are shown in Figure 25.7.5-1.

Category	Mean	5 %	Median	95 %
CDF	2.73E-07	3.30E-08	1.40E-07	8.45E-07

The mean CDF generated based on the sample size of 100,000 is 8.83E-08 /y.

Graphical results, as well as the uncertainty values (mean, 5 percent, 50 percent, 95 percent) from the UNCERT runs are shown in Figure 25.7.5-2 and Figure 25.7.5-3.

The mean values of CDF, LRF and LERF estimated by UNCERT are higher than point estimate value using the logic model. Some significant cutsets include multiple basic events with the same TYPE CODE related to digital components. The State Of Knowledge Correlation (SOKC) related to digital components would cause the differences.

In addition, uncertainty analysis for each fission product release category has been performed as shown in Table 25.7.5-1.

Category	Mean	5 %	Median	95 %
LRF	8.83E-08	6.70E-09	3.10E-08	2.81E-07
LERF	8.68E-08	6.21E-09	2.91E-08	2.77E-07

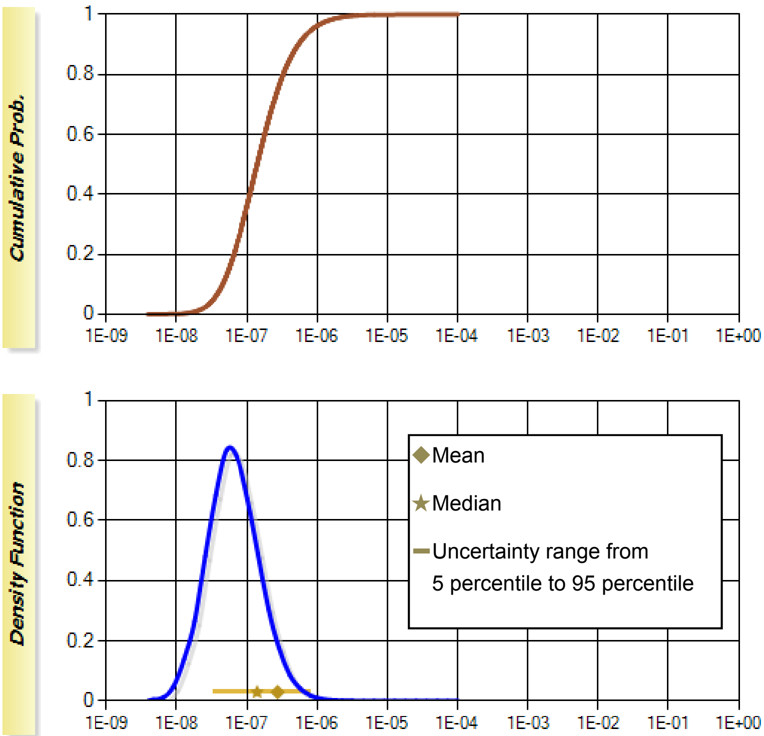

Figure 25.7.5-1 CDF Uncertainty Analysis

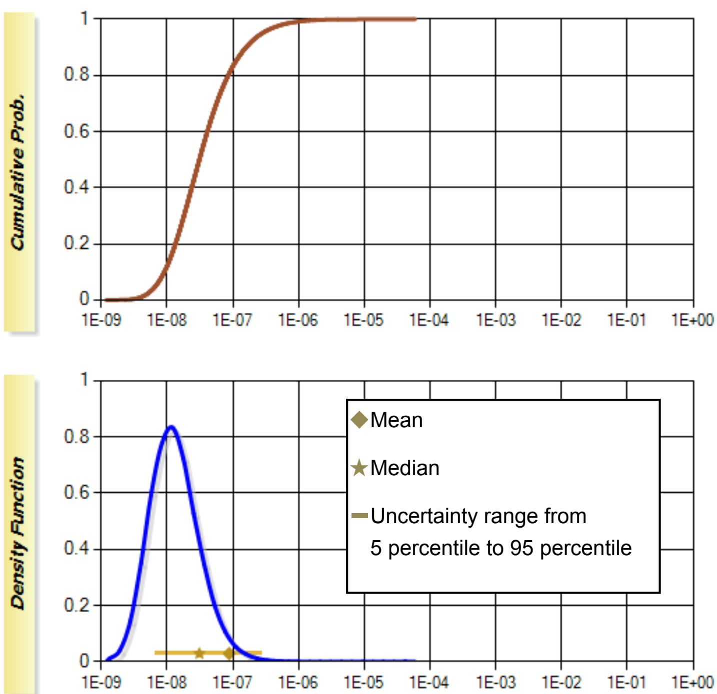

Figure 25.7.5-2 LRF Uncertainty Analysis

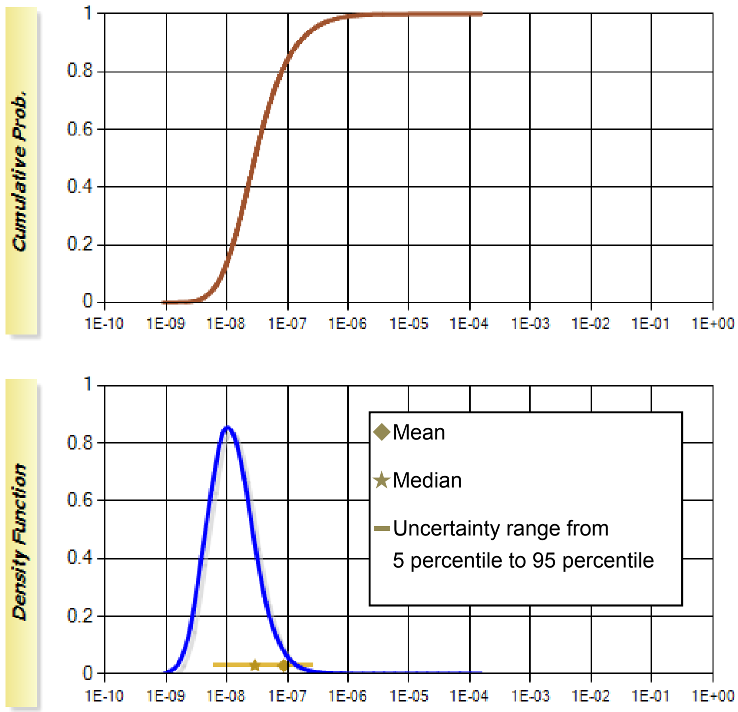

Figure 25.7.5-3 LERF Uncertainty Analysis

Table 25.7.5-1 Uncertainty Analysis for Level 2 Release Category Frequency

No.	Release Category	Mean	5 %	Median	95 %
1	Containment Leakage (KV and KP_I, II and III)	6.49E-08	1.66E-09	1.48E-08	1.75E-07
2	Containment Venting (VV and VP_I, II and III)	8.04E-10	1.74E-11	2.42E-10	3.22E-09
3	Filtered Containment Venting (FVV and FVP_I, II and III)	1.58E-07	1.34E-08	7.94E-08	5.05E-07
4	Early Containment Failure (ALL_IV_except_BYPASS)	2.45E-08	5.08E-10	5.97E-09	8.84E-08
5-1	Late Containment Failure (OP/OT_I)	4.28E-09	3.27E-11	8.30E-10	1.47E-08
5-2	Late Containment Failure (OP/OT_II)	3.47E-11	1.95E-13	5.39E-12	1.25E-10
5-3	Late Containment Failure (ALL_V_except_BYPASS)	2.38E-10	7.07E-12	7.73E-11	8.64E-10
5-4	Late Containment Failure (ALL_VI_except_BYPASS)	3.14E-09	6.62E-11	6.80E-10	1.08E-08
6	Late Containment Failure (OP/OT_PS_I, II and III)	3.63E-09	7.53E-11	9.33E-10	1.32E-08
7-1	In-vessel Fuel-Coolant Interaction (RE_I)	1.09E-12	1.90E-15	9.23E-14	3.60E-12
7-2	In-vessel Fuel-Coolant Interaction (RE_II)	7.65E-14	7.26E-18	2.10E-15	1.98E-13
8-1	Ex-vessel Fuel-Coolant Interaction (PE_I)	1.58E-10	2.06E-12	3.56E-11	5.97E-10
8-2	Ex-vessel Fuel-Coolant Interaction (PE_II)	1.13E-11	4.07E-14	1.44E-12	4.02E-11
9	Direct Containment Heating (DCH_II)	1.55E-11	1.78E-14	1.15E-12	5.29E-11
10-1	PCV Isolation Failure (PCI_I)	5.57E-11	2.59E-12	2.66E-11	1.95E-10
10-2	PCV Isolation Failure (PCI_II)	2.94E-11	5.48E-13	8.65E-12	1.15E-10
11-1	Molten Core Concrete Interaction (CCI_I)	1.73E-09	4.57E-14	1.13E-10	6.40E-09
11-2	Molten Core Concrete Interaction (CCI_II)	7.01E-12	9.45E-17	2.69E-13	2.33E-11
12	RPV Rupture (ALL_IX_except BYPASS and RR and RR_LD_AE)	1.19E-08	1.20E-10	1.63E-09	2.75E-08
13	Containment Bypass (ALL VII and BYPASS_except III)	5.03E-08	2.67E-09	1.19E-08	1.44E-07
14	S/P Bypass (ALL_VIII_except_BYPASS)	9.95E-11	5.09E-13	1.90E-11	3.70E-10
15	Direct Debris Interaction (DDI_II)	3.64E-09	5.45E-11	8.78E-10	1.38E-08
16	Long term SBO (Containment Failure w/o Spray _III)	3.42E-09	5.62E-13	1.14E-10	1.05E-08

25.7.5.2 Sensitivity Analysis

(1) Identification of Sensitivity Analysis Case

Modelling uncertainty was identified from three sources.

- Source of modelling uncertainty 1: Assumptions list
- Source of modelling uncertainty 2: Facts and Observations (F&Os) from peer review
- Source of modelling uncertainty 3: Analysis review

Following the identification of modelling uncertainties, sensitivity study analysis cases were grouped into the following safety functions:

- Reactivity control,
- Core cooling,
- Containment heat removal,
- Hydrogen control,
- Containment, and
- Others.

The sensitivity study cases identified are summarised in Table 25.7.5-2.

Table 25.7.5-2 Summary of Sensitivity Analysis Cases (1/7)

No.	Safety Function	Sensitivity Analysis Case	Purpose of Sensitivity Analysis
1	Reactivity control	Failure of all RIPs resulting in RCPB failure (Class: AC)	To confirm the risk impact if failure of all RIPs does not cause RCPB failure during an ATWS event.
2		Opening all SRVs with failures of RPS and RPT to prevent RCPB failure (Class: AC)	To confirm the risk impact if opening all SRVs with failures of RPV and RPT prevents RCPB failure.
3		ARI initiated by L2 signal	To confirm the risk impact if ARI initiated by L2 signal prevents core damage in the event of TM (loss of condenser heat sink). In addition, combined sensitivity analysis for No.2 and No.3 are performed.
4		Plant scram with failures of support systems	To confirm the risk impact if the following support system initiators are treated as transient initiating event. - Loss of Class 1 DC - Loss of Class 1 AC - Loss of Class 3 AC - Loss of HNCW - Loss of RCW/RSW - CCF of Class 1 controller - CCF of Class 3 controller - Loss of Instrument or Control Air System
5		RPV low pressure scenario during ATWS	To confirm the risk impact if RPV depressurisation and low pressure injection are credited during ATWS.
6		CCF of control rod drive	To confirm the risk impact if CCF among CRD between RPS/ARI and FMCRD run-in is considered.

Table 25.7.5-2 Summary of Sensitivity Analysis Cases (2/7)

No.	Safety Function	Sensitivity Analysis Case	Purpose of Sensitivity Analysis
7	Core cooling	Success criteria in RPV rupture	To confirm the risk impact if core cooling by HPCF is credited to prevent core damage following in RPV rupture.
8		Success criteria for RCIC in LOCA, Stuck Open of Relief Valve (SORV) and Inadvertent Open of Relief Valve (IORV)	To confirm the risk impact if core cooling by RCIC is credited until RPV pressure decreases below the level where RCIC can operate.
9		RCIC operability under loss of Class 1 AC power	To confirm the risk impact if RCIC is operable in 4 hours or 14 hours under loss of Class 1 AC power.
10		Failure location of RCPB due to consequential LOCA	To compare Conditional Core Damage Probability (CCDP) for different break locations as shown below. Representative location: FW-A Other location: One Main Steamline (MSL), SRV Inlets, FW-B, RHR A, B, C Suction and CUW mid-vessel suction. They have the same impact on the systems.
11		HPCF accident operability	To confirm the risk impact if HPCF is operable when S/P temperature is over 171 °C.
12		ECCS suction strainer	The same basic events are applied to suction strainer plugging in the LOCA events and non-LOCA events. To confirm the importance if the basic events are divided into LOCA events and non-LOCA events.
13		Test and Maintenance (T&M) for ADS	To confirm the risk impact if the system level T&M for ADS is divided into three events based on the logic level (ADS, T-ADS and D-ADS).
14		HPCF after recovery of offsite power	To confirm the risk impact if damaged core injection by HPCF is credited to prevent RPV breach in TB, TBP and TBU after offsite power recovery.
15		Addition of HPIN failure	To confirm the risk impact if one of the HPIN design options is adopted for ADS, RDCF and manual depressurisation.

Table 25.7.5-2 Summary of Sensitivity Analysis Cases (3/7)

No.	Safety Function	Sensitivity Analysis Case	Purpose of Sensitivity Analysis
16	Core cooling	Multiple demand for RPV water level control	To confirm the risk impact if multiple demands on valves, relay and so on for RPV water level control between L1.5 (L1) to L8 is considered.
17		Failure of safety relief valve opening	To confirm the risk impact assuming the RPV remains high pressure given failure to open SRVs, no injection credited.
18		SRV success criteria for RDCF and LPFL	To confirm the risk impact if diverse ADS is credited for LPFL.
19		RPV depressurisation for FLSR during SBO	To confirm the impact if manual SRVs opening using nitrogen cylinder is credited during SBO.
20		Spurious ADS/MSIVs and loss of ECCS	To confirm the impact if inadvertent ADS opening/MSIVs closure due to software CCF occurs with loss of RPS/ECCS.
21		CCF between ADS and RDCF	To confirm the risk impact if CCF among SRVs for ADS and RDCF is considered.
22		Additional Credit of MUWC/CRD in the HCTL sequences	To confirm the impact if the MUWC or CRD is credited in the HCTL sequences.
23		In-Vessel Recovery	To confirm the risk impact if In-Vessel Recovery is not credited and CRD is credited to prevent RPV breach failure in the Level 2 PSA.
24		SRV thermal seizure during severe accident	To confirm the risk impact if SRV thermal seizure (stuck open) is considered in Level 2 PSA.

Table 25.7.5-2 Summary of Sensitivity Analysis Cases (4/7)

No.	Safety Function	Sensitivity Analysis Case	Purpose of Sensitivity Analysis
25	Containment heat removal / Containment	W/W venting which is unavailable during a LOCA below TAF	To confirm the risk impact if wetwell venting is not credited in LOCA below Top of Active Fuel (TAF). That is, remaining venting line is the filtered venting from drywell and impact under degrade venting function is assessed by this sensitivity analysis.
26		Adverse effect on equipment outside reactor building following containment failure at a different location	To confirm the risk impact if systems outside the reactor building, e.g., feedwater system, condensate system and FLSS are not credited. FLSS is located in the B/B. However reactor water level transmitter for FLSS is located in R/B B1F. As the bounding sensitivity analysis, credit of FLSS is not taken.
27		Adverse effect on equipment inside reactor building following containment failure	To confirm the risk impact if systems inside the reactor building are credited. CDF of TW and TW-LP is multiplied by 0.1 as conditional probability of the adverse effect (i.e., ECCS survives).
28		Adverse effect on equipment inside reactor building following ISLOCA/BOC	To confirm the risk impact if systems inside the reactor building are not credited. Temporary use of ECCS in non-BOC/ISLOCA division, e.g., 10 min, before FLSS is initiated is still credited.
29		Phenomenological probability	To confirm the risk impact if phenomenological probabilities obtained from analytical approach are used.
30		The effect of one SRV tailpipe break	To confirm the risk impact if the effects of one SRV tailpipe break is explicitly distinguished in event trees.
31		SRV tailpipe break in LOCA	To confirm the risk impact if SRV tailpipe integrity is examined in LOCAs.
32		PCV venting at 1 Pd	To confirm the impact on source term if PCV venting is conducted at 1 Pd.
33		Failure size of containment bypass	To confirm the impact on source term if the location of containment bypass is changed from RHR suction line to MS line.

Table 25.7.5-2 Summary of Sensitivity Analysis Cases (5/7)

No.	Safety Function	Sensitivity Analysis Case	Purpose of Sensitivity Analysis
34	Containment heat removal / Containment	Lower set point of COPS	To confirm the risk impact if the COPS setpoint is lower.
35		Reactor Well Injection	To confirm the risk impact if reactor well injection is credited in the Level 2 PSA.
36		Vacuum Breakers Failure in Level 2 PSA	To confirm the risk impact if vacuum breakers failure (fail to open and stuck open) is considered in the Level 2 PSA.
37	Hydrogen control	De-inerted operation	To confirm the risk impact if the probability of de-inerted operation increases from 0.003 to 0.01 considering unplanned outages.
38	Others	Water shortage in CST during BOC (sampling line break)	To confirm the risk impact if HPCF and RCIC are not credited in BOC sampling line break.
39		Water shortage in case of min flow from CST to S/P	To confirm the risk impact if HPCF and RCIC are not credited when the minimum flow valve is open.
40		Environmental effect on operators in MCR following the loss of MCR HVAC	To confirm the risk impact if uncertainty upper bounds (95 percent) of Human Event Probabilities (HEPs) are used in the event of loss of MCR HVAC.
41		Heat Capacity Temperature Limit (HCTL)	HCTL was conservatively assessed. This conservative HCTL leads shorter time margin for operator action. If detailed assessment is performed, more realistic HCTL may be set and may provide longer time margin for operators. To confirm the risk impact on the TW-LP if uncertainty lower bounds (5 percent) of all Human Event Probabilities (HEPs) are used.
42		Dependency of Multiple Human Failure Events (HFEs)	Cognition errors (___-HFE-CC-CG, ___-HFE-LT-CG, ___-HFE-CC-L2, ___-HFE-DC-L2) are modelled as HEP dependency. To confirm the risk impact if four cognition errors are changed to the one cognition errors (complete dependency) and deleted (zero dependency).

Table 25.7.5-2 Summary of Sensitivity Analysis Cases (6/7)

No.	Safety Function	Sensitivity Analysis Case	Purpose of Sensitivity Analysis
43	Others	Manual BOC isolation	To confirm the risk impact if manual isolation is credited following a BOC.
44		Recovery of offsite power in Level 2 PSA	In Level 2 PSA, offsite power recovery is credited in TB, TBP and TBU. To confirm the risk impact if credit of offsite power recovery is taken in other plant damage states after loss of offsite power.
45		Time reliability correlations for HEP calculation	To confirm the risk impact if time reliability correlations is used for HEP calculation on risk significant HEP.
46		Manual operation for reactivity control	To confirm the risk impact if manual operations for reactivity control such as manual scram and SLC initiation are credited.
47		RPV water level control at LOCA/BOC/ISLOCA	To confirm the risk impact if automatic RPV water level control by HPCF is not credited at BOC/ISLOCA. In addition, wetwell venting is not credited due to submergence of wetwell venting line given failure of FLSS to control water level.
48		Manual initiation of HPCF-C local cooling	In the PSA model, only HPCF-C hardwired logic is modelled based on the design information on PSA freeze date. To confirm the risk impact if the hardwired logic of HPCF-C local cooling is credited for low time margin event when a loss of HVAC does not cause system failure immediately.
49		Chronological position of FLSS in HCTL event tree	To confirm the risk impact if the priority order is changed to HPCF, LPFL, CS and FLSS.
50		PCV venting at high containment temperature	To confirm the risk impact if PCV venting at high containment temperature is credited during a LOCA.
51		Monthly test for HPCF and LPFL injection valves	To confirm risk impact if the monthly tests for HPCF and LPFL injection valves are not conducted during plant operation but on a refuelling outage test interval.

Table 25.7.5-2 Summary of Sensitivity Analysis Cases (7/7)

No.	Safety Function	Sensitivity Analysis Case	Purpose of Sensitivity Analysis
52	Others	Pipe break probability	To change pipe break probability based on NUREG/CR-5862 [Ref-25.55]. This is used for investigation of the potential impact of pipe break frequency to consider NUREG/CR-5862. Also, this is used for investigation of potential benefit of enhancing pressure capability pipe to support ALARP studies.
53		Manual close of spurious opened MOV at ISLOCA	To credit operator actions to isolate the ISLOCA (H1) given that 10 percent of MOV spurious operation is recoverable.
54		Addition of CCF between EDG and BBG	To add the CCF between EDG and BBG and confirm RAW of the CCF.
55		Detection rate by self-testing for C&I digital systems	To confirm the risk impact if the detection rate by self-testing is changed to 80 percent and 95 percent.
56		Failure rates associated with Class 1 Platform	To confirm the risk impact of digital C&I component reliability.
57		Failure rate for sea water system	To confirm the impact if the failure rate for sea water system is doubled.
58		Depletion of battery during non-SBO	To confirm the impact if the battery is not credited in non-SBO condition.
59		Updated HRA report (revision E)	To confirm the risk impact to reflect human error probabilities and dependency treatment as described in HRA report revision E.
60		EDG, BBG and DAG mission time considering LOOP duration	To confirm the risk impact if the EDG, DAG and BBG mission time is appropriately considered considering LOOP duration.
61		LOCA inside containment using a modified apportioning method	To confirm the risk impact if the small and medium LOCAs resulting from larger diameter pipes are considered.

(2) Results of Sensitivity Analysis

Sensitivity Analyses have been performed for each key safety function using quantitative and/or qualitative methods. The sensitivity study results were compared to the results of the current baseline PSA to determine if further study of a topic was required. Any sensitivity cases that vary by more than 20 percent in either CDF or LRF from the baseline case was further evaluated or justified.

The risk measures for benchmarking the sensitivities included:

- Core Damage Frequency (CDF), Large Release Frequency (LRF) and importance measures from internal events at power Level 1 and Level 2 models.
- CDF or LRF differences calculated as the normalised difference between the results from the sensitivity and the baseline model,

$$Difference = {}^{(Sensitivity - Baseline)}/_{Baseline}$$

Table 25.7.5-3 through Table 25.7.5-7 show the sensitivity results for each safety function.

Table 25.7.5-3 Sensitivity Analysis Results for Reactivity Control

No.	Sensitivity Analysis Case	Difference		Remarks
		CDF	LRF	
1	Failure of all RIPs resulting in RCPB failure (Class: AC)	0.0 %	0.0 %	Minimal impact on the PSA results. Plant Damage State (PDS) with the failure of all RIPs is changed from AC to TC. Both PDSs result in over 10 percent of CsI release fraction though the fraction in AC is more than that in TC.
2	Opening all SRVs with failures of RPS and RPT to prevent RCPB failure (Class: AC)	-1.0 %	-3.6 %	Small impact on the PSA results.
3	ARI initiated by L2 signal	-4.9 %	-17 %	Small impact on the PSA results.
	Combination case of No.2 and No.3	-5.9 %	-21 %	Significant impact on the PSA results.
4	Plant scram with failures of support systems	21 %	6.5 %	Significant impact on the PSA results. This sensitivity study may be a bounding case because the support system initiators include one division failure of the support system which would cause the slower affect to the plant.
5	RPV low pressure scenario during ATWS	-0.1 %	0.3 %	Minimal impact on the PSA results.
6	CCF of control rod drive	0.8 %	1.2 %	Small impact on the PSA results.

Table 25.7.5-4 Sensitivity Analysis Results for Core Cooling (1/2)

No.	Sensitivity Analysis Case	Difference		Remarks
		CDF	LRF	
7	Success criteria in RPV rupture	-4.0 %	-14 %	Small impact on the PSA results.
8	Success criteria for RCIC in LOCA, Stuck Open of Relief Valve (SORV) and Inadvertent Open of Relief Valve (IORV)	-1.8 %	-1 to 0 %	Small impact on the PSA results. *3: The small difference of CDF would not significantly impact on LRF because CDF of each PDS is not significantly changed.
9	RCIC operability under loss of Class 1 AC power (RCIC runs for 4 hours)	4.7 %	-0.7 %	Case1: Small impact on the PSA results. Case2: Small impact on the PSA results.
	RCIC operability under loss of Class 1 AC power (RCIC runs for 14 hours)	-2.5 %	0.1 %	
10	Failure location of RCPB due to consequential LOCA	-0.2 %	-1.4 %	Minimal impact on the PSA results.
11	HPCF accident operability	-2.5 %	-8.8 %	Small impact on the PSA results.
12	ECCS suction strainer	0.0 %	0.0 %	Minimal impact on the PSA results.
13	Test and Maintenance (T&M) for ADS	-6.8 %	-0.2 %	Small impact on the PSA results.
14	HPCF after recovery of offsite power	-	0.0 %	Minimal impact on the PSA results based on the total LRF with the recovery of offsite power in the base case.
15	Addition of HPIN failure (Case1: Current RDCF Design)	16 %	0.0 % %	Small impact on the PSA results. This analysis is used for the reference because the purpose is to confirm the risk impact of new RDCF design (Case 2).
	Addition of HPIN failure (Case2: RDCF Design Option suggested by PSA)	0.0 %	0.0 %	Minimal impact on the PSA results, compared with the base case (HPIN is not required for the RPV depressurisation). When the results are compared with Case 1, the CDF is reduced by 16 percent.
16	Multiple demand for RPV water level control	0.3 %	0.1 %	Minimal impact on the PSA results.

Table 25.7.5-4 Sensitivity Analysis Results for Core Cooling (2/2)

No.	Sensitivity Analysis Case	Difference		Remarks
		CDF	LRF	
17	Failure of safety relief valve opening	0.3 %	-1.3 %	Small impact on the PSA results.
18	SRV success criteria for RDCF and LPFL	-1.9 %	0.0 %	Small impact on the PSA results.
19	RPV depressurisation for FLSR during SBO	3.1 %	3.1 %	Small impact on the PSA results.
20	Spurious ADS/MSIVs and loss of ECCS	0.0 %	0.2 %	Minimal impact on the PSA results.
21	CCF between ADS and RDCF	2.7 %	0.2 %	Small impact on the PSA results.
22	Additional Credit of MUWC/CRD in the HCTL sequences	-25 %	-0.2 %	Significant impact on the PSA results.
23	In-Vessel Recovery	-	9.8 %	Small impact on the PSA results.
24	SRV thermal seizure during severe accident	-	0.0 %	Minimal impact on the PSA results.

Table 25.7.5-5 Sensitivity Analysis Results for Containment Heat Removal / Containment (1/2)

No.	Sensitivity Analysis Case	Difference		Remarks
		CDF	LRF	
25	W/W venting which is unavailable during a LOCA below TAF	- (No impact)	0.0 %	Small impact on the PSA results.
26	Adverse effect on equipment outside reactor building following containment failure at a different location	3.2 %	11 %	Small impact on the PSA results. This adverse effect results in core damage and Fission Product is released because containment fails prior to core damage. Therefore LRF is calculated by adding the increment of CDF.
27	Adverse effect on equipment inside reactor building following containment failure	-2.6 %	-9.0 %	Small impact on the PSA results.
28	Adverse effect on equipment inside reactor building following ISLOCA/BOC	5.2 %	18 %	Small impact on the PSA results.
29	Phenomenological probability	- (No impact)	-4.1 %	Small impact on the PSA results.
30	The effect of one SRV tailpipe break	0.0 %	- (judged small)	Minimal impact on the PSA results.
31	SRV tailpipe break in LOCA	0.0 %	- (judged small)	Minimal impact on the PSA results.
32	PCV venting at 1 Pd	- (No impact)	- (No impact)	Confirmed the impact on source term. The release fraction of CsI significantly increased from 4.4E-08 to 2.5E-03. However, the release rate is still lower than 10 percent of CsI release which is the criterion of large release. In addition, FP reduction by filter is not considered in this analysis. Therefore the assumption would not affect the PSA results. (Only MAAP evaluation)

Table 25.7.5-5 Sensitivity Analysis Results for Containment Heat Removal / Containment (2/2)

No.	Sensitivity Analysis Case	Difference		Remarks
		CDF	LRF	
33	Failure size of containment bypass	- (No impact)	- (No impact)	Confirmed the impact on source term. The release fraction of CsI reduced from 0.90 to 0.69. However, the release rate is still higher than 10 percent of CsI release which is the criterion of large release. Therefore the assumption would not affect the PSA results. (Only MAAP evaluation)
34	Lower set point of COPS	-1.0 %	-0.2 %	Small impact on the PSA results.
35	Reactor Well Injection	-	0.0 %	Minimal impact on the PSA results.
36	Vacuum Breakers Failure in Level 2 PSA	-	0.0 %	Minimal impact on the PSA results.

Table 25.7.5-6 Sensitivity Analysis Results for Hydrogen Control

No.	Sensitivity Analysis Case	Difference		Remarks
		CDF	LRF	
37	De-inerted operation	-	1.5 %	Small impact on the PSA results.

Table 25.7.5-7 Sensitivity Analysis Results for Others (1/2)

No.	Sensitivity Analysis Case	Difference		Remarks
		CDF	LRF	
38	Water shortage in CST during BOC (sampling line break)	0.3 %	1.0 %	Small impact on the PSA results.
39	Water shortage in case of min flow from CST to S/P	2.7 %	9.6 %	Small impact on the PSA results.
40	Environmental effect on operators in MCR following the loss of MCR HVAC	0.0 %	0.0 %	Minimal impact on the PSA results.
41	Heat Capacity Temperature Limit (HCTL)	-2 %	-0.5 %	Small impact on the PSA results based on the discussion on F-V importance.
42	Dependency of Multiple Human Failure Events (HFEs) (Complete Dependency)	7.3 %	4.7 %	Small impact on the PSA results.
	Dependency of Multiple Human Failure Events (HFEs) (Independent Dependency)	-0.3 %	-1.6 %	Small impact on the PSA results.
43	Manual BOC isolation	Judged small	-0.1 %	Minimal impact on the PSA results based on the comparison of LOCA CCDP.
44	Recovery of offsite power in Level 2 PSA	- (No impact)	-1.3 %	Minimal impact on the PSA results.
45	Time reliability correlations for HEP calculation	-6.6 %	-0.9 %	Small impact on the PSA results.
46	Manual operation for reactivity control	-6.8 %	-23 %	Significant impact on the PSA results. Failure of manual operation for reactivity control results in containment failure prior to core damage. Therefore LRF is calculated by adding the decrement of CDF.
47	RPV water level control at LOCA/BOC/ISLOCA	1.5 %	5.7 %	Small impact on CDF and LRF.
48	Manual initiation of HPCF-C local cooling	-20 %	6.6 %	Significant impact on CDF and small impact on LRF.
49	Chronological position of FLSS in HCTL event tree	5 %	20 %	Small impact on CDF. Significant impact on LRF. Further analysis or ALARP is be necessary.
50	PCV venting at high containment temperature	- (No impact)	-2.0 %	Small impact on the PSA results.
51	Monthly test for HPCF and LPFL injection valves	0.0 %	0.0 %	Minimal impact on the PSA results.
52	Pipe break probability	-2.2 %	-7.8 %	Small impact on the PSA results.

Table 25.7.5-7 Sensitivity Analysis Results for Others (2/2)

No.	Sensitivity Analysis Case	Difference		Remarks
		CDF	LRF	
53	Manual close of spurious opened MOV at ISLOCA	-0.2 %	-0.9 %	Minimal impact on the PSA results.
54	Addition of CCF between EDG and BBG	4.2 %	2.5 %	Small impact on the PSA results.
55	Detection rate by self-testing for C&I digital systems (Case 1: undetectable rate is 5 percent)	-5.6 %	-14 %	Small impact on the PSA results.
	Detection rate by self-testing for C&I digital systems (Case 2: undetectable rate is 20 percent)	24 %	73 %	Significant impact on the PSA results.
56	Failure rates associated with Class 1 Platform	-6.1 %	-11 %	Small impact on the PSA results.
57	Failure rate for sea water system	7.3 %	0.2 %	Small impact on the PSA results.
58	Depletion of battery during non-SBO	19 %	2.4 %	Small impact on the PSA results.
59	Updated HRA report (revision E)	5.4 %	19 %	Small impact on the PSA results.
60	EDG, BBG and DAG mission time considering LOOP duration	-8.8 %	-0.4 %	Small impact on the PSA results.
61	LOCA inside containment using a modified apportioning method	-0.3 to 0.0 %	-0.4 to -0.2 %	Minimal impact on the PSA results. (impacts depending on assumptions)

(3) Insights of Sensitivity Analysis and Uncertainty Analysis

The most significant insights are shown below, which were based on the results of sensitivity and uncertainty analyses with the internal events at power PSA models. The risk contributions from external events have not been included in the sensitivity and uncertainty analyses yet.

- Overall, it was found that risk impacts by most of the sensitivity cases are not significant except the four cases described below.

- Individual sensitivity cases with CDF or LRF values that vary from the baseline cases by roughly 20 percent or more were identified as significant, which are shown in Table 25.7.5-8 below. Further assessments are considered for these cases.

Table 25.7.5-8 Sensitivity Results that vary from the Base Case by 20 % or more

No.	Sensitivity Analysis Case	Difference		Remarks
		CDF	LRF	
4	Plant scram with failures of support systems	21 %	6.5 %	Significant impact on the PSA results. This sensitivity study may be a bounding case because the support system initiators include one division failure of the support system which would cause the slower affect to the plant.
22	Additional Credit of MUWC/CRD in the HCTL sequences	-25 %	-0.2 %	Significant impact on the PSA results.
46	Manual operation for reactivity control	-6.8 %	-23 %	Small impact on the PSA results. Failure of manual operation for reactivity control results in containment failure prior to core damage. Therefore LRF is calculated by adding the decrement of CDF.
48	Manual initiation of HPCF-C local cooling	-20 %	6.6 %	Significant impact on CDF. Small impact on LRF.
49	Chronological position of FLSS in HCTL event tree	5 %	20 %	Small impact on CDF and significant impact on LRF. Further analysis or ALARP is necessary.
55	Detection rate by self-testing for C&I digital systems (Case 2: undetectable rate is 20 percent)	24 %	73 %	Significant impact on the PSA results.

- Sensitivity case No.4: When support system failure occurs, operators manually shut down the reactor prior to a plant scram. It is found that manual operator action may be difficult due to a reactor trip in a short term when the CCF of Class 3 controller or Class 1 DC occurs simultaneously. The risk impact on CDF is small (2 percent increase) if a plant trip occurs after the initiating events.

- In sensitivity case No.22, the decrease of the CDF is 25 percent and it is found that the additional credit of MUWC is important at HCTL-based depressurisation. That is why loss of RCW/RSW has relatively significant impact on CDF and MUWC pumps do not depend on RCW/RSW.

- Sensitivity case No.46 identified that the credit of manual scram and manual initiation of SLC has large impact on LRF. In this analysis, 0.1 of human error probability is assumed. During ATWS, time margin for reactivity control is generally short. Therefore, the success criteria analyses for manual scram and manual initiation of SLC and justification for the feasibility of the actions are required if the operations are credited.

- In sensitivity case No.48, the decrease of CDF is almost 20 percent if manual initiation of HPCF-C local cooling by hardwired system is credited in all initiating events including the large LOCA. The room heat up calculation is necessary to justify the feasibility of the action if the manual operation is credited.

- Sensitivity case No.49 identified that the prioritisation of the injection systems is important after HCTL-based depressurisation. In the sensitivity analysis, the event trees prioritising HPCF was treated and the sensitivity due to the order of the water injection system was evaluated. If HPCF is successful after RPV depressurisation, FLSR is assumed not to be credited. In the base case, FLSR is not credited under the following scenarios:
  - Operators acknowledge that the water injection system was unable to be expectable with Station Blackout and loss of ultimate heat sink.
  - There is a time allowance of more than 8 hours with the water injection operated by RCIC.

  The procedure for FLSR would be required to conduct the refinement after GDA.

- In sensitivity case No.55, the CDF and LRF increase by 24 percent and 73 percent, respectively. However, this is not an issue for the base case due to the following arguments.
  - Except OLU and LD

    Since platform is under development, the diagnostic rates shown in Topic Report on Class 1 Platform [Ref-25.125] are different from those used in PSA. The diagnostic rates of the modules (except for OLU and LD) in Topic Report on Class 1 Platform are better than that used in the PSA. Therefore, the current PSA uses a bounding value for the modules except OLU and LD.
  - OLU and LD

    The diagnostic rate of LD is actually considered for only the function to re-open the MSIVs. Since the function to re-open the MSIVs is credited in very limited accident sequences, e.g., non-isolation type transients followed by loss of RHR and intact Power Conversion System, no cutset includes the basic events of LD for re-opening MSIVs even with the truncation value 1E-17 /y (convergence was confirmed with this truncation). Even if the diagnostic rate of LD is lower than that used in the PSA, this does not impact the PSA result.

- The mean values generated by uncertainty analysis are CDF of 1.34E-07 /y and LRF of 3.19E-8 /y. The results of internal events at power PSA show that the mean value of LRF is below target 7 and 8 BSO of 1E-6 /y.

- Sixty-one sensitivity analyses have been performed individually. Table 25.7.5-8 shows the sensitivity cases where impact of CDF and/or LRF is significant. If necessary, the further sensitivity analyses for those cases were performed to reduce the conservativeness to understand the impact on CDF and LRF more explicitly.

25.7.5.3 Level 3 PSA Sensitivity Analysis

(1) Non-plant related parameters

A number of decisions were made in respect of the most appropriate non-source term input parameters and modelling assumptions for use in the Level 3 PSA, based on a wide ranging literature review and a number of sensitivity analyses. The decisions made as a result of the sensitivity analysis are summarised below:

- A representative source term phasing scheme is adopted, conservatively accounting for radionuclide decay and in-growth at the start of each release phase for both deterministic and probabilistic calculations.
- Choice of Pasquill Stability Category D and a wind speed of 5m/s is representative of the meteorological dataset for the Wylfa Newydd site and is used for deterministic calculations. A constant wind direction is assumed, aligned with the long dimension of the building as this results in a narrower plume and represents the bounding case in terms of wind direction.
- The choice of 0mm/h rainfall is the most common rainfall rate and is used in the main deterministic calculations; however, where the dose from the 3 main pathways is < 1Sv, a second calculation assuming the average (non-zero) rainfall rate is performed to ensure the robustness of the dose band allocation.
- Probabilistic calculations use sampling of the meteorological datasets and so represent the spectrum of meteorological conditions.
- The choice of deposition velocities in both deterministic and probabilistic calculations are the same, and are reasonable estimates for the UK as a whole. In respect of the dry deposition velocity for elemental iodine, where a wide spread of values are postulated in the literature, a best-estimate value is generally used; however, in the deterministic calculations where the dose from the 3 main pathways is < 1 Sv, a second calculation is performed assuming the higher DBA elemental iodine deposition velocity for the ingestion pathway to test the robustness of the dose band allocation.
- Generic representative person parameters are adopted, as recommended by the UK expert body (Public Health England). This is deemed appropriate for the GDA, where minimal data specific to the site is used. Location factors are not used for the cloud gamma and inhalation pathways in the deterministic calculations but they are used for the probabilistic calculations. A ground gamma integration time of 50 years with the location factor representative of normal behaviour is used in both deterministic and probabilistic calculations. The sensitivity analysis for the deterministic calculations showed that there is typically a small reduction in dose for an integration time of 50 years with a location factor of 0.19 compared to an integration time of 1 year with an indoor occupancy of 0. A second deterministic calculation with an integration time of 1 year and an indoor occupancy of 0 is performed for release categories with a dose < 1 Sv to assess whether the alternative assumption would result in any changes in the facility dose band allocation.
- Assessments against facility dose bands model adults as the representative person. In respect of age at exposure, the total doses (i.e. sum of inhalation, cloud gamma and ground gamma) for infants were shown to be less than a factor of 2 larger than those for adults. Additionally, adult, child and infant doses in the first year including the ingestion pathway are reported for release categories with a dose < 1 Sv. This allows an assessment of the confidence to which a release category can be assigned to the lower facility dose bands.
- The pragmatic decision was made to continue to use the population file provided by PHE, based on 2001 census data, as no readily applicable alternative was available for GDA. The impact of

altering the population data file was considered for both long term notional fatalities (a generic 10 percent uplift in each grid element) and early fatal health effects (by introducing a conservatively high number of people in the first distance band (0 to 800m)).

- A 10 percent increase in population at all grid points (representing population growth between 2001 and 2014) resulted in increases in the number of notional late fatalities of between 0 percent and 6 percent, depending on release category. Adding a significant population within the first distance band (0 to 800m) resulted in an increase in the number of short term health effects leading to fatality of about 5 percent of the added population for the most severe release category.

These decisions were based on the release category definitions and source terms from the December 2014 version of the Level 2 PSA. The changes to the SAA representative sequences do not affect these parameter value choices as they are either independent of the actual source term (representative person parameters, meteorological parameters) or the sensitivity calculations, on which the parameter value choice was partly based, considered a wide range of radionuclide mixes which equally represent the updated SAA.

(2) Plant related parameters

A number of decisions were made in respect of the most appropriate source term related input parameters and modelling assumptions for use in the Level 3 PSA, based on studies for the Level 3 PSA development and a number of sensitivity analyses. The decisions made as a result of the sensitivity analysis are summarised below:

- The basic releases to the environment are predicted using the MAAP code [Chapter 26 of PCSR Rev.C] for the representative SAA sequences for each release category. The sensitivity analysis for the SAA is reported elsewhere.
- As discussed in Section 25.6.2.1, the Level 3 PSA is based on two treatments of iodine releases which are intended to 'bracket' the likely conditions for UK ABWR. These are the SAA Base Case, which assumes all iodine is released as CsI (as in the supporting MAAP analysis), and the Modified Case which includes addition of elemental iodine (I_2) and organic iodine (CH_3I) to the Base Case release. The reported results for the IEAP are for the more conservative Modified Cases. The additional iodine component is available for release from the onset of core melt, is assumed to be released in proportion to the noble gases and is based on the NUREG-1465 methodology [Ref-25.73]. This aspect will be updated as enhanced iodine chemistry models for UK ABWR severe accidents are developed. The results of the sensitivity calculations for the Base Case and Modified Case show:
    - For the deterministic calculations, the assessment against the lower facility dose bands is affected by the volatile iodine in the Modified Cases, most significantly for the venting release categories (P2 and P3).
    - The conditional individual risks for most release categories are not significantly affected by the volatile iodine in the Modified Cases. The increase in conditional individual risk is typically < 10 percent, with larger factors seen only for release categories P2, P6, P8 and P10-1. Release category P2 has proportionally the greatest increase in both source term and conditional individual risk as the venting route is unfiltered; however, release category P2 has a low frequency.
    - Release category P2 is the most directly affected scenario in terms of the conditional societal risk. The combination of the change in representative sequence (from a case with drywell sprays to one without drywell sprays) and addition of volatile iodine in the Modified Case has increased the conditional societal consequences, such that this is currently assessed as being above the Target 9 threshold.

- Release phases that are wholly released via the Standby Gas Treatment System (SGTS) or Filtered Containment Venting System (FCVS) are identified and modelled as stack releases in the dispersion calculations. Particulate filter Decontamination Factors (DFs) for the SGTS are modelled in MAAP 4.07 whilst particulate filter DFs for the FCVS and all iodine DFs are applied by post-processing the release fraction values. A number of previous studies and sensitivity studies have considered the effect of varying the DFs for the FCVS and SGTS, and also the effect of removing the filtration factors completely. The results of these sensitivity studies can be summarised as follows:
    - The assessment against the lower facility dose bands is affected by the assumption of SGTS filtration for the leakage release category P1; however, the impact is limited by the residual dose from noble gas releases. A significantly higher conditional individual risk is seen for release category P1 when the effect of the SGTS filter is removed in the Modified Case.
    - The assessment against facility dose bands for the filtered venting release category P3 is not affected by the filtration factors assumed for the FCVS *per se*, as long as there is some reasonable filtration factor. If filtration is removed, the consequences are the same as for release category P2. Similarly, a higher conditional individual risk is seen for release category P3 when the effect of the FCVS filter is removed in the Modified Case.
    - Overall, these effects have no impact on the individual risk for the IEAP event group.
- Phases that are not entirely released via the SGTS or vent path are conservatively modelled as releases into the building wake, even if a portion of the release may be via the stack. Any thermal energy in the plume causes the material discharged to rise above its original release point. A release energy of 0 MW is assumed for all dose calculations conducted to date for the Level 3 PSA, which is considered to be conservative. For ‘D5’ weather, energy greater than 34 MW is required for the plume to break free of the building wake for a release from the building. In this condition, the subsequent plume dispersion behaves more like an elevated release, resulting in lower doses close to the site and the point of maximum dose moves away from the site. To reach the boundary layer height energy of greater than 100 MW is required for both a building release and a stack release.

(3) Offsite countermeasures

(a) Interventions for the ingestion pathway

For the deterministic and probabilistic calculations, there is an assumption of minimal offsite countermeasures. This effectively means that countermeasures are only applied to the ingestion pathway and are restricted to interventions for food contaminated to above the European Council Food Intervention Levels (CFILs) [Ref-25.72]. The impact of such restrictions was considered in sensitivity studies as follows:

- Ingestion is not included as a pathway in the main deterministic calculations, as it would be expected to be significantly curtailed by implementation of food restrictions. In the additional calculations, performed where the dose from the 3 main pathways is < 1Sv, ingestion doses in the absence of any food restrictions are calculated in the most conservative manner, i.e. assuming all the food is produced and consumed at the location that results in highest dose. The ingestion rates are based on the mean consumption rates for each age group given in NRPB-W41 [Ref-25.63]. The sensitivity studies showed that the contribution from ingestion is most substantial for younger people and particularly for infants, where increases in the total dose of up to a factor of 6 were seen for some release categories. This was attributed to the higher consumption (proportionally to total consumption) of dairy products (primarily milk) assumed for children and infants.

- Probabilistic calculations include ingestion as a pathway using the FARMLAND model in PC COSYMA; however, ingestion of foodstuffs above the intervention levels [Ref-25.72] is not included in the main calculations. The sensitivity analysis showed relatively little increase in the individual risk when ingestion doses are limited by the CFILs, compared to neglecting this pathway; however, the individual risk was found to increase by between a factor of two and six, depending on the release category, in the absence of food restrictions. It was also seen that the inclusion of ingestion doses, even with food intervention levels modelled, can have a significant effect on the number of long term notional fatalities in the population even if the impact on individual risk is small. These long term notional fatalities arise from consumption of food at below intervention level values which is deemed safe to eat. This is a significant finding as these fatalities result from:
  - PC COSYMA calculates societal effects from ingestion by calculating the collective dose from all food that is not restricted and applying a dose / risk factor.
  - The Linear No Threshold (LNT) assumption that any dose, no matter how small, contributes to the risk of late fatal health effects is inherent in the dose / risk factor. Therefore, small doses from foodstuffs at below the intervention levels also contribute to the predicted numbers of notional late fatalities.

(b) Potential impact of other offsite countermeasures

For some release categories there is the potential to implement early offsite countermeasures (i.e. sheltering, issue of stable iodine tablets and/or evacuation) to avert some of the dose to members of the public. Early countermeasures are not claimed in either the individual risk or societal risk calculations for the GDA. Late countermeasures other than food interventions, i.e. relocation and decontamination, are also not claimed in either the individual risk or societal risk calculations.

The potential impact of offsite countermeasures was considered in sensitivity studies. Three Level 3 PSA cases (P2, P5-1 and P13) from the IEAP group were analysed. The three release categories were selected as they result in a range of early and late conditional individual risks and early and late societal health effects. In addition to controls on consumption for the foods predicted to have activity concentrations above the CFILs, the following protective action groups were considered:

- Sheltering alone, sheltering and administration of stable iodine tablets, sheltering followed by evacuation (in the first 24 hours), and
- Sheltering followed by dose based relocation in the long term. Resettlement following relocation was not considered in these sensitivity studies.

Due to the modelling in PC COSYMA, the first group of actions mainly affects short term fatal health effects and the second group affects only late notional (stochastic) fatal health effects.

The criteria used to impose the offsite protective actions are derived from the IAEA generic criteria for protective actions and other response actions in emergency exposure situations to reduce the risk of stochastic effects. The UK specific Emergency Reference Levels were also used in some sensitivity cases.

The results of the protective actions sensitivity studies are:

- Protective Actions to avert short term dose
  - Protective actions to avert short term dose, particularly evacuation, can significantly reduce the individual risk of early fatal health effects in the region where they are imposed. Thus, they can also have a significant impact on the numbers of early fatal health effects in the region where they are imposed.

- Protective actions to avert short term dose have no noticeable impact on the numbers of late notional fatal health effects in the population as a whole, unless the region where they are imposed is large.

- Protective Actions to avert long term dose
  - Protective actions to avert long term dose, particularly relocation, can significantly reduce the individual risk of late notional fatal health effects in the region where they are imposed.
  - Protective actions to avert long term dose can have a significant impact on the numbers of late notional fatal health effects in the region where they are imposed.
  - Protective actions to avert long term dose can have an impact on the numbers of late notional fatal health effects in the population as a whole, particularly where the region they are imposed is large.

The overall findings of the protective actions sensitivity study are:

- Protective actions could have a significant impact on the number of fatal societal health effects, both early fatal health effects and late (stochastic) fatal health effects in the region where they are imposed.
- The number of predicted health effects in the UK population is unlikely to reduce sufficiently to have an impact on the societal risk assessment (Target 9), using the current representative SAA sequences.
- Some benefit could be seen in the individual risk assessment (Target 7); however, these are not claimed as they would be difficult to justify for the GDA.

25.7.6 Insights from Assessment

25.7.6.1 Level 1 Insights

The quantification results provide the following insights:

- Transients have a large contribution (46 percent) to the IEAP CDF. Two thirds of the CDF by transients comes from LOOP events because of the necessity to restart important support systems, e.g. RCW, HVAC, and this depends on the digital control system. LOOP with failure of the digital control system (Software CCFs, CPU CCFs) results in SBO for the Class 1 systems.
- Manual shutdowns (including support system initiators) have a considerable contribution (39 percent) to the IEAP CDF. More than one third of the CDF by manual shutdowns comes from loss of RCW/RSW because CCFs of all the RCW/RSW divisions are modelled.
- About one third of the CDF by LOCAs comes from RPV rupture. Another one third comes from BOC events. The remaining one third comes from LOCAs inside containment and ISLOCAs.
- TQUV is the most significant Accident Class: about 60 percent of the IEAP CDF. This accident class includes various scenarios that are not short term, e.g. failure of low pressure injection after operation of RCIC or operation of the HPCF for 3 hours or more.
- FLSS and BBG have F-V values of more than 50 percent and 40 percent (system level), respectively. This is a notable insight and may raise the need to improve the reliability of the systems and to ensure adequate test and maintenance.
- EDG also has high system level importance (nearly 40 percent) similar to BBG. Prevention of Station Blackout is very important for the UK ABWR.

- HVAC has a F-V value of more than 30 percent because of the significance of failure in a wide range of systems.
- FLSR is credited primarily for the severe accident mitigation. However, this also contributes significantly to the accident mitigation before severe accident.
- Digital C&I system has a F-V value of more than 40 percent (system level). This implies the importance of digital C&I system reliability. But it should be noted that conservative software CCF significantly contributes to that high F-V value.
- Software CCFs (HVAC, RPS, RHR_RCW) have large contributions to the IEAP CDF because of the significance of failure in a wide range of systems. It should be noted that the CCF probability assigned is conservative which leads to a higher significance.
- Test and maintenance unavailabilities of less redundant systems, e.g. FLSS, BBG have higher importance than the test and maintenance unavailabilities of more redundant systems, e.g. ECCS, EDG.
- Failure to align FLSR (expected to dominate the FLSR unavailability) has the highest contribution to the CDF among the post-initiator HFEs. This is due to the high failure probability assigned (0.3) and claiming FLSR as mitigation in a number of risk significant scenarios, e.g. SBO, HCTL.
- Except the FLSR, failures of manual depressurisation have relatively high risk contributions among the post-initiator HFEs.
- Other post-initiator HFEs do not significantly contribute to the CDF. This may be because the UK ABWR has sufficient automation (including passive actuation of COPS).
- Pre-initiator HFEs are not significant risk contributors to the UK ABWR.

25.7.6.2 Level 2 Insights

The quantification results provide the following insights:

- Release category “C”, representing ATWS scenarios, has the highest frequency. Credit for manual scram in case of RPS software failure would reduce the frequency.
- Release category “BYPASS” has the second highest frequency. Credit for manual isolation of PCV would reduce the frequency.
- Release categoryry “RR”, representing RPV rupture, has the third highest frequency. RPV rupture is assumed unmitigable. It is uncertain whether core damage is prevented by automatic actuation of two HPCFs (HPCF spargers above the core) or not. Credit for two HPCFs would reduce the frequency.
- Release category “LTSBO”, long term SBO, has the fourth highest frequency. When the RPV fails due to the failure of core cooling, D/W pressure reaches the twice of design pressure (2Pd) of containment, resulting in containment failure. In this case, PCV venting before 2Pd would be effective to reduce risk.

25.7.6.3 Level 3 Insights

Three release categories are seen to dominate the assessment against the three numerical targets, due to their relatively high frequencies and high conditional consequences:

- IE at Power 4: Early Containment Failure at Power at 2.05E-08 /y of the release frequency,

- IE at Power 13: Containment Bypass at Power at 1.87E-08 /y of the release frequency,
- IE at Power 12: RPV rupture at Power (excessive LOCA) at 1.03E-08 /y of the release frequency.

25.7.7 Key Assumptions and Study Limitations

Assumptions in the internal events at power PSA were made in the development phase. They relate to each aspect of PSA, such as the identification of initiating events, the definition of success criteria, reliability data selection, containment performance, accident progression and so on. In addition, when the level of plant requirements specification limits the detail of the PSA, then appropriate assumptions were developed to allow completion of the PSA.

Starting from assumptions, sensitivity analyses have been performed. Among assumptions in internal events at power PSA, key assumptions, which have comparatively large impact on the result, have been listed from the result of sensitivity analyse.

25.7.7.1 Level 1 key assumptions and limitations

The main assumptions considered in the internal events at power Level 1 PSA are listed below:

IE analysis:

- Seawater intake blockage is identified as one of initiating event candidates, but is not considered in the PSAs for GDA as this IE frequency depends largely on site specific location. This IE is treated at the site licensing stage when considering design changes or countermeasures.
- Some of the support system initiators may cause plant scram in cases where the failure components are not restored. Manual shutdown is not modelled in the event trees of support system initiators because it is assumed that operator can perform manual shutdown.
- RPV rupture is considered as an IE “Excessive LOCA (RPV rupture)” with the frequency of 1.0E-08 /y (converted to 9.0E-09 /y) with an error factor of 62 taken from NUREG-1829 [Ref-25.20]. This is regarded as one of main modelling uncertainties (conservative bias), this IE by itself forms the top LRF cutset.

Success Criteria / Accident Sequence:

- The possibilities for vacuum breaker(s) left open at in a LOCA event (not caused by human error) is treated as a modelling uncertainty.
- BOC / ISLOCA is assumed to cause loss of systems in the same division of R/B. Survivability of R/B systems given BOC / ISLOCA is uncertain.
- When the suppression pool temperature reaches Heat Capacity Temperature Limit (HCTL), operators are required to depress RPV by the accident procedure. The scenario is one of the dominant risk contributors. The following credit for low pressure injection at HCTL-based depressurisation reduces the risk effectively as shown in Table 25.7.5-4 (Case No.22) and Table 25.7.5-7 (Case No.49).
  - RPV injection by MUWC
  - RPV injection by FLSR following HPCF. It is assumed that HPCF pump terminates when S/P temperature reaches 150 ^{o}C.
- The manual operation of HPCF is not credited in LOCAs due to the short time margin to core damage. On the other hand, HPCF-C local cooling by hardwired system would be credited because the room

temperature increases relatively slowly. The additional credit of the manual operation reduces the risk effectively as shown in Table 25.7.5-7 (Case No.48).

<u>Data Analysis:</u>

- Generic Data: Industry average data is used for all component failure and CCF events

  Due to the lack of available plant specific data for a plant that is still in the design phase, generic data has been used. It is assumed that the industry average data (mainly from the US) is applicable to the UK ABWR. This is a reasonable assumption given the fact that BWRs are operating in the US. The industry averaged values may be conservative as newly designed components are likely to have a higher reliability than components in currently operating plants. This introduces a high level of uncertainty into the model. Once the plant is operational, plant specific data may be collected and applied to the model with a Bayesian update process. Parametric uncertainty analysis was performed to understand the impact from the uncertainty for each basic event.

- Digital C&I failure rates:

  There are model uncertainties associated with modelling digital systems, such as those related to determining the failure modes of these systems and components and assigning appropriate data. Values provided by the manufacturer have been used, with large uncertainty distributions. Class 2 and 3 has been modelled at a very high level only. Additional CCFs for Class 1 software failures have been included. It is assumed that the only source of software CCF is due to the functional specification requirements.

  A self-detection rate of 90 percent is used. The values from the manufacturer are likely to be conservative, as is the bounding value used for the software CCF. It is also likely that the detection rate will be higher than 90 percent, so this is also a conservative assumption. This modelling introduces a high level of uncertainty into the model.

<u>Human Reliability Analysis:</u>

- Manual operations for reactivity control are not credited in this PSA. The additional credit of the operations reduces risk for the failure of reactivity control effectively as shown in No.46 of Table 25.7.5-7 although the time available is very short.

25.7.7.2 Level 2 key assumptions and limitations

The main assumptions considered in the internal events at power Level 2 PSA are listed below:

<u>Containment Performance Analysis:</u>

- When RPV failure occurs with high pressure, the access tunnel hatch and the vacuum breakers are assumed to fail due to overtemperature (Direct Debris Interaction). This is the conservative assumption because radiation from very high temperature corium (1,000 K) is assumed in the assessment. When water injection to the lower drywell is successful prior to RPV breach, a large amount of steam is generated and the temperature of the corium deposited on the lower drywell structures decreases. As the result, radiation from the corium dramatically decreases, and the hatches and the vacuum breakers remain intact. On the other hand, it is conservatively assumed that the direct debris interaction causes the containment failure at RPV high pressure failure in the Level 2 PSA.

Accident Progression Analysis:

- The PCV failure area due to overpressure or overtemperature is assumed to be 0.068 m^2, which is based on the containment performance analysis. This is a conservative assumption because the gasket of the PCV head flange is assumed to melt away and the gap of PCV head flange is not closed when PCV is depressurised.

Modelling of Level 2 PSA:

- Manual isolation of BOC is not credited. This assumption results in the increase of importance of containment isolation function in BOC events.
- All systems connected to the RPV and PCIS valves inside primary containment are assumed to fail due to an RPV rupture event. This assumption results in the increase of importance of an RPV rupture event. RPV rupture is regarded as one of main modelling uncertainties (conservative bias) because this IE by itself forms the top LRF cutset.

25.7.7.3 Level 3 key assumptions and limitations

The offsite dose assessment for the lower facility dose bands is sensitive to the filtration factors claimed in the Modified Cases for the SGTS and the FCVS. However, the overall assessment against Target 8 is not affected by the filtration factors assumed for the FCVS, as long as there is some reasonable filtration factor for particulates and elemental iodine. Even if the effect of filtration is not credited in either the SGTS or the FCVS, the contribution of internal events leading to fuel melt remains below the BSO in all Target 8 dose bands.

The Level 3 PSA calculations for the GDA are based on assumption of minimal offsite countermeasures; effectively limiting countermeasures to interventions for food contaminated to above the European Council Food Intervention Levels (CFILs) [Ref-25.72]. The SAA predicts sufficient warning time in some of the representative scenarios for implementation of early offsite countermeasures; however, no credit is taken in the Level 3 PSA for reduction in dose afforded by sheltering during the release period, and/or precautionary evacuation prior to release. No other long term countermeasures are modelled, e.g. relocation following the release. Sensitivity studies have shown some benefit could be seen in the individual risk assessment (Target 7) from claiming early offsite countermeasures; however, this benefit is not considered for the GDA. Early and other late offsite countermeasures could also reduce the number of predicted health effects in the UK population but this benefit is unlikely to have a significant impact on the societal risk assessment (Target 9), using the current representative SAA sequences.

25.8 Shutdown PSA

25.8.1 Scope of Shutdown PSA

The scope of the Shutdown PSA includes the reactor core for those periods when the reactor well gate is closed. When the reactor well and Spent Fuel Pool (SFP) gates are open, the scope of the evaluation is expanded to include the risk of damage to fuel stored in the SFP. The SFP is considered separately when the reactor well gate is closed.

Note: as low power conditions were found to introduce no significant initiators unique to low power, low power operations are considered as part of the UK ABWR at Power PSA (see Section 25.7).

25.8.2 Shutdown Plant Operational States

The Shutdown PSA encompasses six Plant Operating States (POSs):

POS S: Transition to reactor cold shutdown

POS S covers the 24 hours after the vacuum break of the main condenser. This POS is characterised by closed RPV/PCV heads, the highest decay heat, and the availability of all the RHR and ECCS divisions (except turbine driven RCIC). The reactor water level is the same as that for at Power operation. The decay heat of the reactor's full core is removed by the shutdown cooling mode of one RHR-A train.

POS A: Transition to reactor disassembly and reactor well gate open with Division 2 in maintenance

POS A is defined as the period from the end of POS S to completion of reactor well flooding and opening the reactor well gate. This POS covers the second and third day during which the reactor is disassembled (opening RPV/PCV top heads), and the reactor water level is raised. The decay heat is still high during this period. Although the actual water level is higher than that in the normal operation, the representative water level for the Shutdown PSA is set as the normal water level. The decay heat of the reactor's full core is removed by the shutdown cooling mode of RHR-A train as in POS S.

For the Level 1 PSA, POS A is represented by a closed RPV head condition because this condition gives more onerous success criteria, e.g. low pressure injection requires RPV depressurisation, reactor well injection is not credited. For source term assessment, POS A is represented by open RPV/PCV heads because this condition would lead to larger potential release.

POS B-1: Full water level in reactor well and gate open with Division 2 in maintenance

POS B-1 is defined as the period from opening the reactor well gate to shutdown of operating RHR-A. The decay heat of the reactor's full core and SFP is removed by the shutdown cooling mode of RHR-A train and FPC-A train. The water inventory is large with the reactor well flooded and the well gate open, so there would be considerable time before reactor coolant boiling even if decay heat removal is lost. During this period, the risk evaluation for the SFP is included with the Shutdown PSA since the SFP and reactor are coupled by the open reactor well gate.

POS B-2: Full water level in reactor well and gate open with Divisions 1 and 3 in maintenance (SFP risk is included in Shutdown PSA due to opened gate)

POS B-2 is defined as the period from shutdown of operating RHR-A to closing the reactor well gate. Plant conditions are the same as POS B-1, and the SFP risk is included with the Shutdown PSA due to coupling through the open reactor well gate. The decay heat of the reactor's full core and SFP is removed by the shutdown cooling mode of RHR-B train and FPC-B train.

NOTE: RHR-B in shutdown cooling mode and its support systems are put in service before the end of POS B-1.

POS C: Transition to closed condition of PCV/RPV heads with Divisions 1 and 3 in maintenance

POS C is defined as the period from closing the reactor well gate through starting drain off of the reactor well to completing the RPV leak test. Inspection and maintenance of equipment may continue in this period. Although the water level is higher than that at normal operation, the representative reactor water level for the Shutdown PSA is set as the normal at Power water level. The decay heat of the reactor's 3/4 full core is removed by the shutdown cooling mode of the RHR-B train. The decay heat in the reactor core is much less than that just after reactor shutdown.

For the Level 1 PSA, POS C is represented by a closed RPV head condition because this condition gives more onerous success criteria, e.g. low pressure injection requires RPV depressurisation, and reactor well injection is not credited. For source term assessment, POS C is represented by open RPV/PCV heads because this would lead to larger potential releases.

POS D: Preparation of plant start-up

POS D is defined as the period from completing the RPV leak test through PCV assembly and leak test to starting control rod (CR) withdrawal for start-up. During this period, inspection and maintenance of equipment for heat removal and makeup will have been completed, so all ECCS divisions except the turbine-driven RCIC, and most other mitigation systems will be available. The decay heat of the reactor's 3/4 full core is removed by the shutdown cooling mode of the RHR-B train.

Maintenance of Backup Building systems (e.g. FLSS, BBG) are performed in POS D. Divisions 1 and 2 would not simultaneously be in maintenance. It is assumed BB Division 1 is in standby and BB Division 2 is in maintenance.

25.8.3 Initiating Events

The main objectives of the IE analysis for internal events shutdown PSA are:

- Investigate and identify a reasonably complete set of the events that potentially lead to core damage,
- Prioritise/group them, and
- Provide estimates for the frequencies of the initiating event groups using information available and associated estimation techniques.

The following approaches/methodologies of IE frequency estimation are recommended in Section 25.6.7 of the AESJ Standard [Ref-25.21].

- Generic data and/or operational experience (if judged applicable), used only for Loss of Offsite Power (LOOP),
- IE frequency used for at Power PSA (if judged applicable),

- Logic model (fault tree analysis, event tree analysis and/or human reliability analysis), e.g., support system initiators, or

- Engineering judgement (e.g., expert elicitation process on RPV rupture frequency).

IE frequency estimation for UK ABWR Shutdown PSA is based on the failure rates of related SSCs and/or human reliability analysis, except for LOOP and RPV rupture.

Each IE of the UK ABWR PSA is quantified on a per calendar year basis. In the Shutdown PSA, the following common approach was taken to derive the initiating event frequency on a per calendar year basis.

- Initiating event frequencies for the at Power condition (POS F) were calculated on a calendar year basis. An availability factor of 0.9 was used to convert any reactor critical year frequencies to calendar year frequencies.

- To avoid omission or double-counting of internal events risk, it is assumed that 0.1 (=1.0-0.9) of a calendar year is in outage. There are two different outage conditions regarding the existence of irradiated fuel in the reactor: (1) Non-full core offload condition where part of the irradiated fuel exists in the reactor (POS S, A, B-1, B-2 and C, D), (2) Full core offload condition where irradiated fuel does not exist in the reactor. The SFP PSA (Section 25.9) assumes that the full core offload condition (POS E) occupies three percent of the calendar year. Therefore, the period covered by the Shutdown PSA (non-full core offload condition) is assumed to be 0.07 (=1.0-0.9-0.03) of a calendar year. Since the operation cycle of the UK ABWR is assumed as 1.5 years, the average period of an outage is 365 days times 1.5 years times 0.07: 38 days.

- For the generic data provided on a reactor critical year basis, (e.g., LOOP) the reactor critical year frequency is multiplied by 0.07 as the "total" IE frequency for the Shutdown PSA. That "total" IE frequency for the Shutdown PSA (representing 38 days per 1.5 years) is further allocated to the relevant POSs as needed using the duration of each POS as a weighting factor. The weighting factors are as follows.

  - POS S: 1 day / 27.5 days = 0.0364

  - POS A: 2 days / 27.5 days = 0.0727

  - POS B-1: 8 days / 27.5 days = 0.291

  - POS B-2: 5 days / 27.5 days = 0.182

  - POS C: 4 days / 27.5 days = 0.146

  - POS D: 7.5 days / 27.5 days = 0.273

NOTE: "27.5 days" does not represent the average duration of an outage per calendar year. The "27.5 days" is used as denominator for only calculating the POS weighting factors.

- For the IE frequencies derived by system fault trees or hand calculation based on hourly component failure rates, the frequency per hour is multiplied by 24 [h/d] times 365 [d/y], and then 0.07. That "total" IE frequency is further allocated to the relevant POSs using the duration of each POS as a weighting factor as previously described.

- For demand-based IEs, (e.g., Leakage during FMCRD inspection (HE)) the probability per demand is multiplied by the number of demands per periodic inspection period divided by the inspection interval to calculate an annual occurrence frequency. The periodic inspection is assumed as every 18 months (1.5 years) for the GDA phase.

The Shutdown IEs are summarised in Table 25.8.3-1. The calculated IE frequencies are summarised in the PSA Summary Report [Ref-25.1]. The individual Shutdown IEs are discussed in the subsections below.

25.8.3.1 Loss of Heat Removal

Loss of operating RHR front line

This IE group includes:

- Loss or spurious failure of operating RHR, and
- Loss or spurious failure of local cooling unit for operating RHR pump room.

The RHR system fault trees are used to quantify this IE group. Division A is considered in POSs S, A and B-1 while the Division B train is considered in POSs B-2, C and D according to the outage schedule defined in Section 25.8.2.

The failures causing loss of the operating RHR train but not causing other dependent failures of mitigation systems are included in the system fault tree, including:

- Spurious isolation signal of RHR pump room temperature high, and
- Failure of RCW function around the RHR heat exchangers (RCW function itself not lost).

Spurious operation of standby ECCS

This IE has following characteristics.

- Standby RHR division is spuriously initiated as LPFL mode at:
  - POS S: Divisions B, C,
  - POS A: Division C, and
  - POS D: Divisions A,C.
- Standby HPCF division is spuriously initiated at:
  - POS S, D: Divisions B, C.

The possible impact is RPV overfill resulting in isolation of the operating RHR train in shutdown cooling mode (due to high RPV pressure) and loss of the spuriously initiated ECCS division(s). In addition, reactor well overfill is assumed to cause loss of Division 1 systems in POS B-1 due to internal flooding in the R/B.

Spurious operation of standby FLSS

This IE has following characteristics.

- FLSS is spuriously initiated at:
  - POS S, A, C: Divisions A, B, and
  - POS D: Division A (NOTE: It is assumed Division B is in maintenance).

The possible impact is RPV overfill resulting in isolation of the operating RHR train in shutdown cooling mode (due to high RPV pressure) and loss of FLSS. In addition, reactor well overfill is assumed to cause loss of Division 1 systems in POS B-1 due to internal flooding in the R/B.

Loss of operating RCW/RSW

The RCW/RSW system fault tree was used to quantify this IE group. Division A is considered in POS S, A and B-1 while Division B is considered in POS B-2, C and D according to the outage schedule defined in Section 25.8.2.

The possible impact is loss of the operating RHR train in shutdown cooling mode as well as failures of systems dependent on the failed RCW.

Loss of R/BEEE/Z HVAC

Loss or spurious failure of Reactor Building Emergency Electrical Equipment Zone HVAC (R/BEEE/Z HVAC) may eventually lead to loss of the Class 1 AC buses in the same division, resulting in the same impact as Loss of Class 1 AC.

The R/BEEE/Z HVAC system fault tree was used to quantify this IE group. Division A is considered in POS S, A and B-1 while Division B is considered in POS B-2, C and D according to the outage schedule defined in Section 25.8.2.

Loss of Hx/B-N HVAC

Loss of Heat Exchanger Building Normal HVAC (Hx/B-N HVAC) may eventually cause loss of TCW and TSW. The impact of this on the mitigation systems is captured in the master fault tree.

In order to cool the Class 1 SSCs (e.g., RCW pumps) in the Heat Exchanger Building, loss of Hx/B-N HVAC results in the automatic start of Heat Exchanger Building Emergency HVAC (Hx/B-E HVAC) and associated HECW (for emergency). If the outdoor temperature is above 25 °C, failure of Hx/B-E HVAC or HECW (for emergency) results in loss of RCW/RSW in the same division, which is consistent with the treatment in the at Power PSA.

The Hx/B-N HVAC system fault tree was used to quantify this IE group.

Loss of Class 1 AC

Loss of Class 1 AC causes a loss of the operating RHR train and its support systems. The possible causes are failures around the M/C or P/C. The impact of this event is loss of the operating RHR train and support systems as well as the mitigation systems dependent on the failed AC division.

The Class 1 AC power supply system fault tree was used to quantify this IE group. Division A is considered in POS S, A and B-1 while Division B is considered in POS B-2, C and D according to the outage schedule defined in Section 25.8.2.

Loss or spurious failure of reactor vessel instrument

Some failure modes of the reactor vessel instrument system may isolate the RHR suction line, resulting in loss of the operating RHR train (as an IE) and also disabling use of the shutdown cooling mode in the standby RHR trains.

CCF of Class 1 DC

Loss of a single division for Class 1 DC does not affect the operating RHR train. However, loss of two or more divisions of Class 1 DC generates a fail-safe isolation signal for the RHR suction line valves.

This IE has following characteristics.

The in-service conditions of Class 1 DC during an outage are assumed for the Shutdown PSA.

- POS S, D: The same as at Power conditions (four DC divisions charged by three MCC divisions).

- POS A, B-1: C-D (Division II) is in maintenance. DC Divisions A, B are charged by MCC Division I. DC Divisions C, D is charged by MCC Division III. DC Divisions B, D is charged using the backup chargers.

- POS B-2, C: AC-C, E (Divisions I, III) are in maintenance. DC Divisions A, B, C, D are charged by MCC Division II. DC Divisions A, C are charged using the backup chargers.

The Class 1 DC power supply system fault trees are used to quantify this IE group.

Failure of switchover of RHR shutdown cooling mode (HE)

The RHR shutdown cooling mode is switched from Division A to Division B at the end of POS B-1. Failure of this switchover due to human error(s) could lead to loss of decay heat removal (RHR-A and RHR-B assumed unavailable).

Loss of a Class 1 AC bus (HE)

It is assumed that the single division Class 1 AC power supply system is lost due to wrong operation. The operation related to the human error is demanded once in POS A where Class 1 AC-D is removed from service. The human error of wrongly removing AC-C instead of AC-D is considered, which results in loss of operating RHR-A (initiator).

25.8.3.2 Loss of Offsite Power

LOOP is treated separately from loss of decay heat removal IEs in order to capture the dependency of various systems. Even if Class 1 AC power supply by an EDG is established after LOOP, restart of the RHR, as well as its support systems is needed.

LOOP (Generic)

The accident sequence of LOOP events in the Shutdown PSA is simpler than that in the at Power PSA, i.e., only one timing of offsite power recovery is considered for each accident sequence in the Shutdown PSA. Therefore, in the Shutdown PSA a non-recovery probability which is derived per the time to fuel damage in each sequence is multiplied to the specific cutsets (loss of all EDGs and BBGs) for that sequence.

Loss of TBNEE/Z HVAC

Loss of Turbine Building Normal Electrical Equipment Zone HVAC (TBNEE/Z HVAC) may eventually lead to a LOOP (with no restoration).

The TBNEE/Z HVAC system fault tree was used to quantify this IE group.

Loss or spurious failure of Class 3 AC

Class 1 M/Cs receive offsite power through the Class 3 M/C. Therefore, Loss of Class 3 M/C leads to LOOP in specific Class 1 Division.

- Loss of Class 3 AC-A1 causes LOOP in Division 1 trips operating RHR-A
- Loss of Class 3 AC-B1 causes LOOP in Division 2 trips operating RHR-B

Impacts on the Class 3 systems are also captured in the master fault tree.

A special purpose IE fault tree was used to quantify the frequency of loss of AC-A1 or AC-B1.

Loss of HNCW

This IE group includes Loss of HNCW as well as loss of its support systems (TCW, TSW) as they have identical impact on the plant. Loss of HNCW eventually causes a LOOP if the outdoor temperature is above 25 °C which is consistent with the treatment in the at Power PSA.

From these assumptions, this IE frequency is calculated considering the conditional probability of outdoor temperature above 25 °C.

The HNCW system fault tree (including TCW and TSW model) was used to quantify this IE group.

25.8.3.3 LOCA above Normal Water Level

When the reactor is not connected with the SFP (POS S, A, C, D), a LOCA above the normal water level does not impact the operating RHR train or draindown to TAF, and therefore is not treated as an IE. When the reactor well is full and connected with the SFP (POS B-1, B-2), this event drains down the SFP resulting in decoupling of the reactor and the SFP, as well as result in a loss of FPC. In this event, the reactor continues to be cooled by the RHR.

LOCA (mechanical) above normal water level

This IE group includes the following IEs.

- LOCA at reactor well drain line
- Dropped load to reactor well
- Dropped load to Steam Dryer, Separator (S/D) pit

LOCA at reactor well drain line:

The reactor well drain line is isolated by manual valve F510. External leakage of this valve is considered representative for frequency quantification. The assumed leak rate was taken from NUREG/CR-6928 2010 [Ref-25.17].

Dropped load to reactor well and Dropped load to S/D pit

For POS B-1 and B-2, none of the loads moved exceed 97 tons. It is assumed that heavy load drop on the reactor well or S/D pit causes a small leak.

The occurrence frequencies of dropped load to reactor well and S/D pit at POS B-1 and POS B-2 are evaluated. These frequencies are modified based on detailed task analysis.

LOCA through MS line plug

This IE group includes following IEs.

- Leakage during MS/PCV inspection (HE)
- Failure of reactor well seal
- LOCA at MS line inside PCV
- LOCA at RCIC steam line inside PCV
- Spurious SRV open
- Spurious MSIV open

Leakage during MS/PCV inspection (HE):

The leak-test of the MS plug is performed and the air pressure of the air-packing is monitored to assure the soundness of the MS line plug. Even if leakage occurs due to functional loss of the air-packing, it does not lead to loss of significant coolant because leakage is limited by the MS plug itself which is pushed to the MS nozzle by the coolant pressure and because the MS nozzle is surrounded by additional U-packing.

Integrity of the water stop function for the PCV bulkhead hatch/manhole flange is maintained by appropriate management of fastening and by visual checking from directly under the drywell side at the point that the hatch is in under the water after starting water filling to the RPV well. Even in the case where the leak occurs after closure/water filling, the hatch and manhole bodies press over the opening part which are pressed by head pressure of RPV well; therefore, the leak size is expected to be “bleeding” level and it will not cause a large amount of loss of primary coolant.

Nevertheless, this event is treated as an IE. This type of leak could occur during POSs B-1, B-2 (full reactor well connected to SFP) if the main steam line plug or PCV bulkhead plate manhole is inadequately attached and operators fail to recognise/isolate the leak.

Failure of reactor well seal:

This IE considers leakage in the PCV bulkhead hutch/manhole flange which is not caused by human errors. Since the cause of leakage is expected to be dominated by inadequate fastening by operators, the frequency of this IE is negligible compared to the “Leakage during MS/PCV inspection (HE)”.

LOCA at MS line inside PCV:

The LOCA frequency is conservatively taken from the frequencies of LOCA groups which include MS lines or MS drain lines in the at Power PSA.

- Large LOCA (MSL, SRV Inlets, FW-B, RHR-A, B, C Suction, CUW mid-vessel Suction) within containment: 7.35E-06 per calendar year for at power (taken from Table 25.4.1-2).

- Small LOCA (CUW Bottom Head Drain and Instrument Taps) within containment: 8.00E-05 per calendar year for at Power (taken from Table 25.4.1-2).

Since the MS lines are plugged during POS B-1 and B-2, this IE requires additional occurrence of leakage from the MS plug. It is assumed that the frequency of leakage from the MS plug is dominated by human error(s) and thus represented by “Leakage during MS/PCV inspection (HE)”.

Therefore, this IE frequency for each POS is the LOCA frequency multiplied by the demand probability of “Leakage during MS/PCV inspection (HE)”. This is negligible compared to the frequency of “Leakage during MS/PCV inspection (HE)” itself.

LOCA at RCIC steam line inside PCV:

The LOCA frequency is conservatively taken from the frequencies of LOCA groups which include the RCIC steam line in the at Power PSA.

- Medium LOCA (RCIC and RPV Head Spray and Instrument Taps) within containment: 2.78E-04 per calendar year for at Power (taken from Table 25.4.1-2).

- Small LOCA (CUW Bottom Head Drain and Instrument Taps) within containment: Already counted in “LOCA at MS line inside PCV”.

Since the MS lines are plugged during POS B-1 and B-2, this IE requires additional occurrence of leakage from the MS plug. It is assumed that the frequency of leakage from the MS plug is dominated by human error(s) and thus represented by “Leakage during MS/PCV inspection (HE)”.

Spurious SRV open:

Since the MS lines are plugged during POS B-1 and B-2, this IE requires additional occurrence of leakage from the MS plug. It is assumed that the frequency of leakage from the MS plug is dominated by human error(s) and thus represented by “Leakage during MS/PCV inspection (HE)”.

Spurious MSIV open:

Since the MS lines are plugged during POS B-1 and B-2, this IE requires additional occurrence of leakage from the MS plug. It is assumed that the frequency of leakage from the MS plug is dominated by human error(s) and thus represented by “Leakage during MS/PCV inspection (HE)”.

25.8.3.4 LOCA between Normal Water Level and TAF

LOCA at feedwater line inside PCV

Since the reactor pressure is low during an outage (except for a few hours during hydrostatic test), a LOCA is very unlikely. But this event is not precluded due to large uncertainty. Leakage from the valves rather than a pipe break would dominate the cause. The impact of this event would be the RPV draindown from the normal water level or full reactor well level to the elevation of the feedwater line. Furthermore, when the RHR-A (SDC) is in operation, this event would cause loss of operating RHR because the RHR-A injection line is connected to one of the feedwater lines.

NOTE: Since each feedwater line is isolated by at least two check valves, the risk from Break Outside Containment (BOC) should be represented by the lines not isolated, i.e., operating RHR and CUW, for which BOCs are explicitly considered.

There is one check valve and one manual valve on each feedwater line inside the PCV.

At low pressure, pipe breaks would not be expected at the high design pressure piping. Additionally, the component leak frequencies are likely conservative since they are calculated based on full system pressure and temperature. As a result, the calculated frequencies are conservative.

NOTE: The manual valve(s) inside the PCV is closed during inspection of the condensate and feedwater system. However, that manual valve(s) is assumed to be always open in order to conservatively estimate the LOCA frequency.

RHR flow diversion

There are potential flow diversions from operating RHR. A special purpose fault tree was used to quantify this IE group.

LOCA at RHR suction line inside PCV

This event is applicable to all POSs. Since the reactor pressure is low during an outage (except for a few hours during hydrostatic testing), a LOCA is very unlikely. But this event is not precluded due to large uncertainty. Leakage from the valves rather than a pipe break would dominate the cause. The impact of this event would be loss of the operating RHR train and the RPV draindown from the normal water level or full reactor well level to the elevation of the RHR suction line.

NOTE: Since the water level is drained to the elevation of the RHR suction line, the function of the SDC mode is lost regardless of the train in which the LOCA occurs.

There is one motor operated valve and one manual valve on each RHR suction line inside the PCV.

At low pressure, pipe breaks would not be expected at the high design pressure piping. Additionally, the component leak frequencies are likely conservative since they are calculated based on full system pressure and temperature.

NOTE: External leak small (1 to 50 gpm, defined in NUREG/CR-6928 [Ref-25.17]) is excluded from the failure modes for LOCA frequency since a very mild plant response is expected.

LOCA at LPFL return line inside PCV

This event is applicable to all POSs. Since the reactor pressure is low during an outage (except for a few hours during hydrostatic testing), a LOCA is very unlikely. But this event is not precluded due to large uncertainty. Leakage from the valves rather than a pipe break would dominate the cause. The impact of this event would be loss of the operating RHR train and the RPV draindown from the normal water level or full reactor well level to the elevation of the LPFL injection nozzles.

NOTE: Since the water level is drained to almost the same elevation as that of the RHR suction line, the function of the SDC mode is lost regardless of the train in which the LOCA occurs.

NOTE: LPFL-A is connected to one of the feedwater lines; therefore, only LPFL-B, C are considered for this IE.

NOTE: Since each LPFL return line is equipped with two check valves, the risk from BOC should be represented by the lines not isolated, i.e., operating RHR and CUW, for which BOCs are explicitly considered.

There is one testable check valve, one air operated valve (for the testable check valve) and one manual valve on each LPFL return line inside the PCV.

At low pressure, pipe breaks would not be expected at the high design pressure piping. Additionally, the component leak frequencies are likely conservative since they are calculated based on full system pressure and temperature.

NOTE: External leak small (1 to 50 gpm) is excluded from the failure modes for LOCA frequency since a very mild plant response is expected.

LOCA at HPCF injection line inside PCV

All POSs are applicable. Since the reactor pressure is low during an outage (except for a few hours during hydrostatic test), a LOCA is very unlikely. But this event is not precluded due to large uncertainty. Leakage from the valves rather than a pipe break would dominate the cause. The impact of this event would be loss of the operating RHR train and the RPV draindown from the normal water level or full reactor well level to the elevation of the HPCF injection line.

NOTE: Since the water level is drained below the elevation of the RHR suction line, the function of the SDC mode is lost.

NOTE: Since the HPCF injection line is isolated by two valves, the risk from BOC should be represented by the lines not isolated, i.e., RHR and CUW, for which BOCs are explicitly considered.

The valve configuration of the HPCF injection line inside the PCV is the same as that of the LPFL return line inside PCV.

At low pressure, pipe breaks would not be expected at the high design pressure piping. Additionally, the component leak frequencies are likely conservative since they are calculated based on full system pressure and temperature. As a result, the calculated frequencies are conservative.

LOCA at RHR suction line outside PCV

Since the isolation valves are normally closed in the RHR trains in standby or out of service, the risk from LOCA at RHR suction line outside PCV is dominated / represented by the "operating" RHR train where the isolation valves are normally open. RHR-A is considered during POS S, A and B-1 while RHR-C is considered during POS B-1, C and D, according to the assumed outage schedule defined in Section 25.8.2. Since the coolant pressure is low, a LOCA is very unlikely. But this event is not precluded due to large uncertainty. Leakage from the valves, pump and heat exchanger rather than pipe break would dominate the cause. The impact of this event would be loss of operating RHR and the decrease in the RPV water level to the elevation of the RHR suction line.

A special purpose fault tree was used to quantify this IE group.

The lowest design pressure and temperature of the RHR system, when used as shutdown cooling mode, are 1.37 MPa and 182 °C, respectively. It is assumed that the coolant pressure and temperature are kept below 0.85 MPa and 52 °C throughout the period covered by Shutdown PSA. Actual operating pressure during an outage is nearly the atmospheric pressure. At the lower pressure, pipe breaks would be unlikely given the higher design pressure. Additionally, the component leak frequencies are likely conservative since they are calculated based on full system pressure and temperature.

For the purpose of comparison, the same approach for BOC frequency estimation as used in the at Power PSA is applied. The frequencies of this IE are calculated based on the EPRI "Pipe Rupture Frequencies for Internal Flooding PRAs" [Ref-25.42]. These conservative frequencies are higher than the frequencies based on the external leaks.

NOTE: The frequencies provided in EPRI "Pipe Rupture Frequencies for Internal Flooding PRAs" include both piping and non-piping contributions.

LOCA at CUW outside PCV

This event is applicable to all POSs. Since the reactor pressure is low during an outage (except for a few hours during hydrostatic testing), a LOCA is very unlikely. But this event is not precluded due to large uncertainty. Leakage from the valves, pumps and heat exchangers rather than a pipe break would dominate the cause.

The frequency of this IE is calculated by the external leak rate from components, e.g., valves, pump, heat exchangers. The special purpose fault tree is used to quantify this IE group.

The operating conditions are as follows.

- POS S, A and D: Two pumps and two non-regenerative heat exchangers are used.

  (NOTE: POS A is included in this group for giving a higher IE frequency than the other group.)

- POS B-1: CUW pump A and non-regenerative heat exchangers A, C are used.

- POS B-2 and C: CUW pump B and non-regenerative heat exchangers B, D are used.

At low pressure, pipe breaks would not be expected at the high design pressure piping. Additionally, the component leak frequencies are likely conservative since they are calculated based on full system pressure and temperature.

Leakage during switchover of RHR shutdown cooling mode (HE)

When the standby RHR train is initiated during switchover to RHR shutdown cooling mode, outflow of reactor coolant may occur if the minimum flow valve for the RHR pump is inadvertently left open.

Leakage during ECCS/RHR system inspection (HE)

This type of leak could happen given multiple human errors. The frequency of this IE is estimated by qualitative task analysis and HEP quantification.

Leakage during CUW system inspection (HE)

This type of leak is similar to the Leakage during ECCS/RHR system inspection. One of the CUW trains (pump, Hx, etc.) is in operation while the other is out of service.

25.8.3.5 LOCAs below TAF

This type of LOCA requires make-up to match both the long term flow through the break and decay heat.

RPV rupture during hydrostatic test

This event is the catastrophic failure of the reactor pressure vessel which could occur at penetrations or welds and lead to RPV failures exceeding a Large LOCA. RPV rupture may directly result in core damage. Since catastrophic failure of the RPV is very unlikely when the reactor pressure is low, this IE is only considered during the hydrostatic test in POS C where the reactor pressure is temporarily raised. Even at high pressure, this event should be still very unlikely but is not precluded due to large uncertainty of frequency and its impact.

LOCA (mechanical) below TAF

The IEs categorised as LOCA below TAF have identical dependency on the mitigation systems. Among them, the following IEs are grouped into “LOCA (mechanical) below TAF inside PCV”.

- Leakage from RIP
- Leakage from FMCRD housing
- Leakage from SRNM
- Leakage from LPRM
- LOCA at CUW inside PCV

That is because the event progression after the occurrence of a LOCA should be essentially the same, i.e., recognising leakage and initiating makeup (isolation not likely). For the success criteria derivation in the event sequence analysis task, the largest leakage rate among these IEs is applied.

Other LOCAs are not grouped due to different dependencies and/or different mitigation measures (including operators’ responses).

Due to the low pressure and temperature during outage (except hydrostatic testing), the risk from this IE group is represented by external leak from valves (below TAF) in the CUW line. According to the CUW P&ID, there are two manual valves below TAF inside the PCV.

NOTE: External leak small (1 to 50 gpm) is excluded from the failure modes for LOCA frequency since a very mild plant response is expected.

Spurious RPV draindown by CUW

During an outage, CUW operates in refuelling mode or RPV spray mode [Chapter 12 of PCSR Rev. C].

The CUW design indicates that inadvertent open or internal leak of both the CUW blowdown flow control valve F022 (AOV) and Low Conductivity Waste system (LCW) side discharge valve F023 (MOV) may result in draindown of RPV water to the LCW. Similarly, inadvertent opening or internal leak of both the CUW blowdown flow control valve F022 (AOV) and S/P side discharge valve F026 (MOV) may result in draindown of RPV water to the S/P.

Potential causes regarding the LCW side discharge valve F023 are:

- Inadvertent open of MOV, and
- Internal leak in MOV.

Potential causes regarding the S/P side discharge valve F026 are:

- Inadvertent open of MOV, and
- Internal leak in MOV.

Potential causes regarding the CUW blowdown flow control valve F022 (AOV) are:

- Inadvertent opening of AOV, and
- Internal leak in AOV.

Opening (as designed) due to inadvertent open of the LCW side discharge valve F023 or the S/P side Leakage during FMCRD inspection (HE)

The risk of a leak is avoided by verifying the Control Rod (CR) coordinates during installation of a CR in the RPV, verifying the CR housing cap installation after CR displacement when removing a FMCRD from the RPV or removing the water stop closing plate installed on the CRD housing of the coordinates before installing a FMCRD in the RPV. In addition, risk is managed by avoiding simultaneous operation of CR movement, FMCRD removal and installation. If a FMCRD or closing plate is removed without a CR in place, a leak from the CRD housing is expected. The pressure head occurring at the CRD outlet is equal to that of the reactor well filled with water.

Leakage during replacement of ICM nozzles (HE)

During Local Power Range Monitor (LPRM) replacement, removal of drain pipe on the pedestal side and in-core nut recovery are scheduled in advance, to be carried out after completing installation of all LPRMs on the operating floor. A team approach was used to minimise the risk of a leak. Team members on the lower drywell operations grating, the operating floor and the control room coordinate using wireless communications to ensure the proper sequence of actions. If a drain pipe is removed before an LPRM is installed, leakage from In-Core Monitor (ICM) housing can occur. This sequence of activities and their consequences are similar to leakage through the FMCRD described above.

Leakage during RIP inspection (HE)

Since RIP maintenance starts by draining the RIP casing, this maintenance operation should not initiate a loss of primary coolant (external leak). However, it is evaluated to assess the risk of human error. A team process, similar to that used for ICM nozzle replacement, is used to minimise the risk of leakage. Leakage from the RIP casing is considered for the case that the closing plate is removed without installing the impeller shaft, or if the primary and secondary seal functions have deteriorated. The sequence of activities and their consequences are almost the same as for leakage from ICM nozzles.

The frequency of this IE is estimated by qualitative task analysis and HEP quantification.

RPV draindown during water level decreasing by CUW (HE)

After closing the pool gate and the RPV hydrostatic test, the reactor water level is decreased back to the normal level using the CUW. RPV draindown could occur if the correct process is not followed and automatic isolation by the Primary Coolant System Isolation System (PCIS) fails.

25.8.3.6 LOCA at SFP

Loss of SFP inventory

Some of the IEs identified for the internal events SFP PSA (see Section 25.9.2) could cause one of the broad IE categories for the shutdown PSA during POS B-1 and POS B-2 in which the SFP risk is analysed together in the shutdown risk analysis. Among them, the IEs not covered by those identified in Subtask 1 in this document are:

- Equipment/crane falling into the SFP,
- Dropped Cask/heavy load drop, and
- Loss of Spent Fuel Pool Liner.

These IEs added from the SFP PSA are grouped as “Loss of SFP inventory”.

Dropped load to SFP, Equipment/crane falling into the SFP, Dropped Cask/heavy load drop.

For POS B-1 and B-2, none of the loads moved exceed 97 tons. It is assumed that heavy load drop (below 97 tons) on the SFP, SFP edge, or reactor building floor cause a small leak (design leak rate) in the SFP.

The occurrence frequencies of dropped load to the SFP, SFP edge and RB floor at POS B-1 and POS B-2 were evaluated. Frequencies are modified based on detailed task analysis.

Loss of Spent Fuel Pool Liner

The frequency is estimated in the SFP PSA. This is also used for the Shutdown PSA by converting to a per calendar year frequency.

LOCA in FPC injection line with failures of countermeasure to siphoning

This IE is treated as large leak (un-mitigatable SFP draindown event) consistent with the SFP PSA (Section 25.9.3.3).

The frequency of this event is calculated in the SFP PSA (Table 25.9.2-1). This is also used for the Shutdown PSA by converting to a per calendar year frequency.

For the purpose of understanding the uncertainty, the IE frequency is checked if the 95 percentile values are used for the individual failure probabilities contributing to this IE.

25.8.3.7 Comparison with Generic Data

The calculated IE frequencies were compared to those in the generic data. The possible reasons for significant differences for some IEs were investigated.

- Loss of R/BEEE/Z HVAC frequency in UK ABWR PSA is higher than the generic data (NUREG/CR-6144) by one order of magnitude. According to the analysis of the IE fault tree for R/BEEE/Z HVAC, more than 80 percent of the IE frequency comes from non-redundant components (i.e., MCC, circuit breaker, damper), although R/BEEE/Z HVAC for each division has redundant fans.

- CCF of the Class 1 DC frequency in the UK ABWR PSA is orders of magnitude smaller than the generic data (EPRI TR-113084). Since the ABWR DC power supply system has strong redundancy/diversity, no single CCF event causes this IE. CCF of batteries and CCF of battery chargers must occur to cause CCF of Class 1 DC.

- The leak event frequency above TAF during open pool gate condition (LOCA (mechanical) above normal water level and LOCA through MS line plug) in the UK ABWR is based on a LOCA at the drain line, heavy load drop, and leakage caused by human error. The frequency is one order of magnitude smaller than the generic value. Generic values might include all possible draindown mechanisms.

- LOCA frequencies in the UK ABWR are based on external leakage from components within the boundary. This frequency is significantly higher than the generic value which is based on WASH-1400 LOCA frequencies. Combination of external leak for all relevant components might be conservative for the purpose of LOCA frequency estimation.

Table 25.8.3-1 Summary of Internal Initiating Events for Shutdown PSA (1/2)

Broad category	Initiating events	Applicable POS	Remarks
Loss of decay heat removal	Loss of operating RHR frontline	All	Initiator fault tree
	Spurious operation of standby ECCS	S, A, B-1, C, D	Initiator fault tree
	Spurious operation of standby FLSS	S, A, B-1, C, D	Initiator fault tree
	Loss of operating RCW/RSW	All	Initiator fault tree
	Loss of R/BEEE/Z HVAC	All	Initiator fault tree
	Loss of Hx/B-N HVAC	All	Initiator fault tree
	Loss of Class 1 AC	All	Initiator fault tree
	Loss or spurious failure of reactor vessel instrument	All	Initiator fault tree
	CCF of Class 1 DC	All	Initiator fault tree
	Failure of switchover of RHR shutdown cooling mode (HE)	B-1	Based on HRA
	Loss of a Class 1 AC bus (HE)	A	Based on HRA
LOOP	Plant centred	All	Based on at Power PSA (section 25.4) Recovery to be considered in event sequence analysis and quantification
	Switchyard centred	All	
	Grid related	All	
	Weather related	All	
	Loss of TBNEE/Z HVAC	All	Initiator fault tree
	Loss of Class 3 AC	All	Initiator fault tree
	Loss of HNCW	All	Initiator fault tree
LOCA above normal water level	LOCA (mechanical) above normal water level	B-1, B-2	Valve leak, Heavy load drop
	LOCA through MS line plug	B-1, B-2	Dominated by “Leakage during MS/PCV inspection (HE)” which is based on HRA

Table 25.8.3-1 Summary of Internal Initiating Events for Shutdown PSA (2/2)

Broad category	Initiating events	Applicable POS	Remarks
LOCA between normal water level and TAF	LOCA at feedwater line inside PCV	All	Based on external leak from components in the boundary
	RHR flow diversion	All	Initiator fault tree
	LOCA at RHR suction line inside PCV	All	Based on external leak from components at the boundary
	LOCA at LPFL return line inside PCV	All	Based on external leak from components at the boundary
	LOCA at HPCF injection line inside PCV	All	Based on external leak from components at the boundary
	LOCA at RHR suction line outside PCV	All	Initiator fault tree
	LOCA at CUW outside PCV	All	Initiator fault tree
	Leakage during switchover of RHR shutdown cooling mode (HE)	B-1	Based on HRA
	Leakage during ECCS/RHR system inspection (HE)	B-2	Based on HRA
	Leakage during CUW system inspection (HE)	B-2	Based on HRA
LOCA below TAF	RPV rupture during hydrostatic test	C	Based on NUREG-1829
	LOCA (mechanical) below TAF inside PCV	All	Based on external leak from components at the boundary
	Spurious RPV draindown by CUW	All	Based on failure(s) of valves
	RPV draindown during water level decreasing by CUW (HE)	C, D	Based on HRA
	Leakage during FMCRD inspection (HE)	B-1	Based on HRA
	Leakage during replacement ICM nozzles (HE)	B-1	Based on HRA
	Leakage during RIP inspection (HE)	B-1	Based on HRA
LOCA at SFP	Loss of SFP inventory	B-1, B-2	Dominated by Heavy Load drop over RB Floor
	LOCA in FPC injection line with failures of countermeasure to siphoning	B-1, B-2	Heady Load Drop Analysis and SFP PSA (Section 25.9,25.11)

25.8.4 Event Sequence Analysis

25.8.4.1 Subtask 1: Definitions of Acceptance Criteria

Fuel cooling

Failure of the fuel cooling function would cause global fuel damage. Therefore, the acceptance criterion is to keep the collapsed water level above the TAF in the reactor and/or SFP.

Reactor coolant pressure boundary protection

Failure to open the necessary number of SRVs in postulated pressurisation events could result in loss of the RCPB, i.e., a consequential LOCA event. A pressure of 120 percent of the RCPB design pressure is chosen as the acceptance criterion (same as the at Power PSA).

Vapour suppression

Vapour suppression function is considered to examine the availability of fission product scrubbing effect by the S/P at the loss of decay heat removal type events for POSs S and D. Therefore, the vapour suppression function focuses on the integrity of the SRV tailpipes. The acceptance criterion is that the integrity of the SRV tailpipes is maintained in both the D/W and W/W airspace.

Containment isolation

The containment isolation function in the Shutdown PSA aims at isolating the normally open lines that penetrate both the RPV and PCV. The acceptance criterion is that the inboard isolation valve or outboard isolation valve is closed and/or kept closed without internal/external leak, which is identical to the success criterion.

25.8.4.2 Subtask 2: Establishment of System Success Criteria

The success criteria analysis applied different criterion to each POS. POS specific event trees and functional fault trees were developed for the Shutdown PSA model. System success criteria were defined for each POS and each initiating event group, based on the system availability (given a specific initiator and POS) and system capacity (e.g., injection rate compared to evaporation rate and/or leak rate, available water inventory against required water inventory).

25.8.4.3 Subtask 3: Derivation of Time Margin for Fuel Cooling and Containment Isolation

Step1: Determination of water inventory

The available water inventory for crediting the time margin depends on the accident phases/scenarios. The overall water inventory is divided into regions as illustrated in Figure 25.8.4-1.

The representative water level used for the calculation of time margins for each POS is assumed to be the lowest planned water level during the POS and represents a model uncertainty for the Shutdown PSA. Therefore, the initial water level during POS A and POS C are set to the same as that during POS S and POS D.

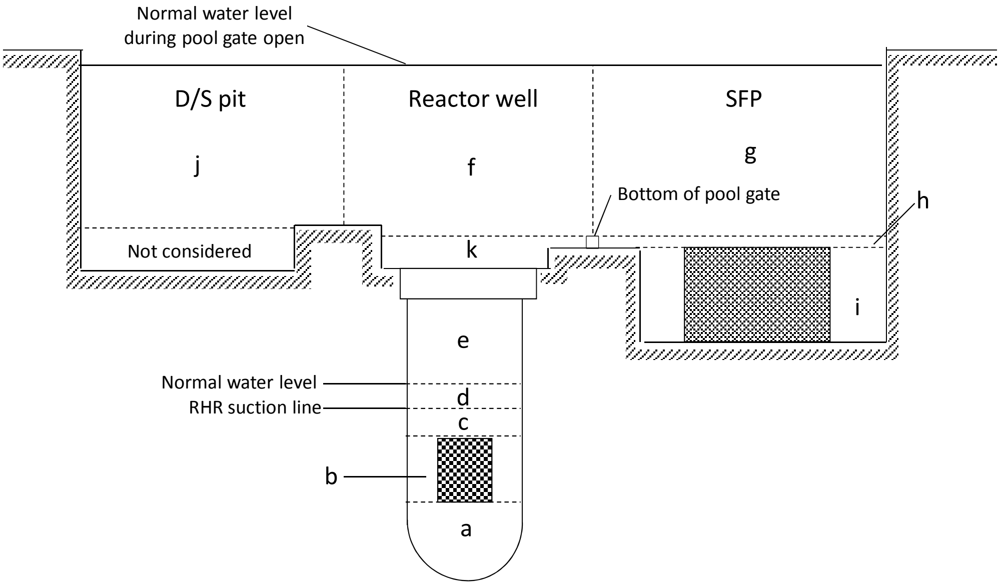

Figure 25.8.4-1 Division of Water Inventory for Time Margin Calculation

Step2: Calculation of time margins

Based on the decay heat level, leak rate and water inventory and accident progression scenario, the various time margins are calculated based on the conditions below:

- Constant decay heat level for each POS throughout accident progression,
- Constant leak rate for each IE throughout accident progression,
- Initial water temperature of 52 °C, and
- Atmospheric pressure throughout accident progression (NOTE: RPV pressurisation when the RPV head is closed is conservatively not considered for the time margin calculation).

If onset of boiling is expected during a specific duration, the evaporations rate is considered throughout that time window as well as leak rate.

25.8.4.4 Subtask 4: Selection of Time Margin for Fuel Cooling and Containment Isolation

Event trees for the UK ABWR Shutdown PSA were developed using CAFTA (Computer Aided Fault Tree Analysis) code, as used in the internal events Level 1 at Power PSA.

25.8.4.5 Subtask 5: Definition of success and non-success end states and key safety functions

Definition of Success End State

In the UK ABWR Shutdown PSA, the success end state is achieved by either of the following end states.

During pool gate closed (POS S, A, C, D):

- Upon loss of decay heat removal events including LOOP:
  - Recovery of heat removal by RHR shutdown cooling mode*

    (*Recovery of mechanically failed RHR is not considered. The impact of this is studied by sensitivity analysis.)
  - Makeup against decay heat in the reactor
- Upon LOCAs:
  - Isolation of a LOCA (for isolatable LOCAs) and makeup against decay heat in the reactor
  - Makeup against decay heat in the reactor if a LOCA location is above TAF
  - Makeup against leak rate and decay heat in the reactor if a LOCA location is below TAF

During RPV pool gate open (POS B-1, B-2)

- Upon loss of decay heat removal events including LOOP:
  - Recovery of heat removal by RHR shutdown cooling mode*

    (*Recovery of mechanically failed RHR is not considered. The impact of this is studied by sensitivity analysis.)
  - Makeup against decay heat in the reactor and SFP
- Upon LOCAs above NWL:
  - Makeup to the SFP against decay heat in the SFP
- Upon other type of LOCAs in the reactor:
  - Isolation before the reactor and SFP decoupled (for isolatable LOCAs)
  - Isolation after loss of operating RHR (for isolatable LOCAs) and makeup against decay heat in the reactor and SFP
  - Makeup against leak rate and decay heat in the reactor and SFP
  - Makeup to the reactor against decay heat in the reactor if a LOCA location is above TAF, and makeup to the SFP against decay heat in the SFP
  - Makeup to the reactor against leak rate and decay heat in the reactor if a LOCA location is below TAF, and makeup to the SFP against decay heat in the SFP

- Upon LOCAs in the SFP
  - Makeup to the SFP against leak rate and decay heat in the SFP

In the base Shutdown PSA, the "Recovery of heat removal by RHR shutdown cooling mode" includes:

- Initiation of the standby RHR division,
- Restart of the operating RHR division after recovery of offsite power or EDG initiation,
- Restart of RHR-A given the IE "Failure of switchover of RHR shutdown cooling mode (HE)", and
- Initiation of standby RHR-C given the IE "Leakage during switchover of RHR shutdown cooling mode (HE)" and successful isolation of minimum flow line.

Reactor and/or SFP boiling for all POSs are assessed against Targets 7 and 8. As an input to this assessment, the success end states are categorised into two groups.

RS : Success sequence by RHR in shutdown mode

RHR shutdown cooling mode is initiated or restarted before the onset of boiling. Therefore, steam is not released from the reactor and SFP.

CS : Success sequence by water makeup in shutdown mode

RHR shutdown cooling mode is failed or not credited. The decay heat is removed by sensible heat and latent heat of the injected water. Therefore, steam is released from the reactor and/or SFP. Since the containment boundary is open for all POSs, this group is commonly applied to all POSs.

Definition of Non-Success End State

Failure to achieve any of the success end states was treated as fuel damage. The accident classes were defined based on the following elements:

- RPV head condition,
- Containment condition,
- Pressure at fuel uncovery (when RPV head is closed),
- Mechanism of fuel uncovery (boil-off or draindown),
- Place of fuel damage (reactor and/or SFP), and
- Any other conditions regarding safety functions considered.

Failure to achieve any of the above success end states was treated as fuel damage. The accident classes were defined as the following. Table 25.8.4-1 shows the summary.

POSs S and D: RPV head is assumed to be closed for both the PSA and the source term assessment. RPV pressure is explicitly distinguished. Since the scrubbing effect by the S/P is important for the source term, its availability is explicitly distinguished. Five Accident Classes are defined for POSs S and D as shown in Table 25.8.4-1.

I Fuel damage due to boil-off at low pressure (with S/P bypass)

II Fuel damage due to boil-off at high pressure (with S/P bypass)

III Fuel damage due to draindown

IV Fuel damage due to boil-off at low pressure (S/P available for scrubbing)

V Fuel damage due to boil-off at high pressure (S/P available for scrubbing)

NOTE: If fuel damage is caused by both draindown and boil-off, i.e., onset of boiling before fuel uncovery at LOCA below TAF (not isolated), the end state is categorised as I or II because the source term is expected to be larger than that for III.

POSs A and C: The RPV head is assumed to be closed for the PSA and to be open for the source term assessment. Although the RPV pressure could be high in some PSA sequences, it is always treated as low pressure. Also, the scrubbing effect by the S/P is always treated as unavailable in the source term analysis [Ref-25.77]. Two Accident Classes are defined for POSs A and C as shown in Table 25.8.4-1.

I Fuel damage due to boil-off at low pressure (with S/P bypass)

III Fuel damage due to draindown

NOTE: If fuel damage is caused by both draindown and boil-off, i.e., onset of boiling before fuel uncovery at LOCA below TAF (not isolated), the end state is categorised as I because the source term is expected to be larger than that for III.

POSs B-1 and B-2: The RPV head is assumed to be open for both the PSA and the source term assessment. The RPV pressure is always low and the scrubbing effect by the S//P is always unavailable. Location of fuel damage is explicitly distinguished. Five Accident Classes are defined for POSs B-1 and B-2 as shown in Table 25.8.4-1.

VI Fuel damage in both the reactor and the SFP due to boil-off

VII Fuel damage in the reactor only due to boil-off

VIII Fuel damage in the SFP only due to boil-off

IX Fuel damage in the SFP only due to draindown

NOTE: According to the time margin analysis, boiling would always be initiated before fuel damage in the reactor and/or SFP at the IE “Loss of SFP inventory”. Accident Class VIII is assigned to the fuel damage end state of this IE because the source term only considering boil-off is larger than that which considers boil-off and small leak. Accident Class IX is only applied to the un-mitigatable LOCA at the SFP: A LOCA in the FPC injection line with failures of countermeasure to siphoning.

Table 25.8.4-1 Accident Classes

Relevant POS	Class	RPV head condition for PSA	RPV head condition for source term assessment	RPV pressure for source term assessment	Containment condition	Mechanism of fuel uncovery[3]	Location of fuel damage	Availability of S/P for FP scrubbing
S, D	I	Close	Close	Low	Not intact (de-inert)	Boil-off	Reactor	No
A, C	I	Close	Open	Low[1]	Not intact (de-inert)	Boil-off	Reactor	No
S, D	II	Close	Close	High	Not intact (de-inert)	Boil-off	Reactor	No
S, D	III	Close	Close	Low	Not intact (de-inert)	Draindown	Reactor	No
A, C	III	Close	Open	Low[1]	Not intact (de-inert)	Draindown	Reactor	No
S, D	IV	Close	Close	Low	Not intact (de-inert)	Boil-off	Reactor	Yes[2]
S, D	V	Close	Close	High	Not intact (de-inert)	Boil-off	Reactor	Yes[2]
B-1, B-2	VI	Open	Open	Low	Not intact (de-inert)	Boil-off	Reactor/SFP	No
B-1, B-2	VII	Open	Open	Low	Not intact (de-inert)	Boil-off	Reactor	No
B-1, B-2	VIII	Open	Open	Low	Not intact (de-inert)	Boil-off	SFP	No
B-1, B-2	IX	Open	Open	Low	Not intact (de-inert)	Draindown[4]	SFP	No

[1] High pressure condition is possible in the PSA end states, but low pressure condition is always assumed for the source term assessment due to the open RPV head.
[2] Given success of the vapour suppression function and the containment isolation function at non-LOCA events (including LOCAs successfully isolated).
[3] If boiling is expected at the moment of fuel uncovery, “Boil-off” is assigned regardless of coincident draindown.
[4] Only applied to “LOCA in FPC injection line with failures of countermeasure to siphoning”.

25.8.4.6 Subtask 6: Event tree development

This subtask is the main part of event tree development. The role of an event tree is a graphical expression of the IE group, success criteria and sequence end point. A full list and description of the event trees used in the Shutdown PSA is given in the Topic Report on Internal Events Shutdown Level 1 PSA [Ref-25.75].

Dependency treatment

The IE to system dependencies summarised as the Dependency Matrices in the Dependency Notebook, are substantiated in the system/functional fault trees and/or event trees. Specific dependencies explicitly substantiated in the event tree development are listed below.

- Systems/trains out of service during specific POSs are removed from the headings.
- In the LOCA event trees where isolation of RHR is expected, the RHR heading is removed.
- LPFL is basically not credited during POSs B-1 and B-2 due to reduced water level in the S/P. However, an LPFL heading is created for the LOCA event trees where leaked coolant would return to the S/P, e.g., LOCAs inside the PCV.
- Change of success criteria with/without isolation of a LOCA is reflected in the event tree structure.
- Failure of RCPB protection is assumed to cause a consequential LOCA which directly causes fuel damage.
- Failure of the vapour suppression function or the containment isolation function is examined at the end of the loss of decay heat removal event trees for POSs S and D in order to distinguish the accident class with/without crediting the fission product scrubbing effect by the S/P.
- Some event trees are used for multiple IE groups. Dependencies from the initiators are automatically captured by the integration of initiator fault trees (or initiator basic events) and mitigating system fault trees.

Condition setting

Flag files are not used for the UK ABWR Shutdown PSA unlike the at Power PSA. Instead, IE basic events are put to various positions of system/functional fault trees to control the available SSCs, initiation signals, operator actions and so on.

25.8.5 System Analysis

The System Analysis for the SFP PSA follows the same basic procedure as that of the Internal Events at Power PSA, see Sections 25.4 and 25.7.

25.8.5.1 Identification of Fault Trees for Internal Events Shutdown PSA

System fault trees were derived by means of the following steps:

(1) Identifying the safety functions,

(2) Identifying the front-line safety systems,

(3) Identifying support systems of the above front-line systems, and

(4) Developing linkage of fault trees.

First, the safety functions that should be performed to prevent fuel damage were identified. The following safety functions were considered in the Level 1 Shutdown PSA:

- Fuel cooling: All POSs
- RCPB protection: POSs S, A, C and D upon loss of decay heat removal type IEs
- Vapour suppression: POSs S and D upon loss of decay heat removal type IEs
- Containment isolation: All POSs (for isolating BOC), POSs S and D (for avoiding S/P bypass) upon loss of decay heat removal type IEs

Second, the front-line safety systems that perform the required safety functions were identified. Front-line safety systems credited in sequence analysis task are shown in Table 25.8.5-1.

Third, the support systems for front-line safety systems were identified. Operation of the front-line systems needs the support of other systems such as electrical power, component cooling and room cooling.

Finally, the support system fault trees were created for each support system and linked at the component level with each other.

25.8.5.2 List of Fault Trees

A full list and description of the fault trees used in the Shutdown PSA was developed [Ref-25.75].

Table 25.8.5-1 Systems to be Modelled in Internal Events Shutdown PSA

Function	System		System Code
Heat removal	RHR-Shutdown cooling mode		E11
	Fuel Pool Cooling and Clean-up System (FPC)		G41
Water injection	High Pressure Core Flooder System (HPCF)		E22
	Make-up Water Condensate System (MUWC)		P13
	Suppression Pool Clean-up System (SPCU)		G51
	Flooding system of Specific Safety Facility (FLSS)		E71
	Safety Relief Valve (SRV)		B21
	RHR-LPFL mode		E11
	Flooding system of Reactor Building (FLSR)		E72
Support System	Electrical Power Distribution system (R10)	AC Power Supply	-
		- Metal-Clad Switchgear (M/C)	R22
		- Power Centre (P/C)	R23
		- Motor Control Centre (MCC)	R24
		Emergency Diesel Generator system (ED/G)	R43
		Alternative Generator system (A/G)	R44
		DC Power Supply	R42
	Reactor Building Cooling Water System (RCW)		P21
	Reactor Building Service Water System (RSW)		P41
	Emergency Equipment Cooling Water System (EECW)		P27
	Heating Ventilating and Air Conditioning System (HVAC)		U41
	HVAC Normal Cooling Water System (HNCW)		P24
	HVAC Emergency Cooling Water System (HECW)		P25
	I & C (including digital system)		A32
	Turbine Building Cooling Water System (TCW)		P22
	Instrument Air System (IA)		P52
	Station Service Air System (SA)		P51

25.8.6 Level 1 Quantification and Results

This section summarises the quantification of the Internal Events Shutdown Level 1 PSA for the UK ABWR. The contents include the quantification results including the FDF, dominant POSs, initiators, accident sequences and accident classes, and basic event importance measures.

25.8.6.1 Model Results Summary

The UK ABWR PSA model consists of event trees and fault trees that are quantified using a fault tree linking process. The calculation of the total FDF is performed as a single top gate.

As a result of the quantification, the total FDF during Shutdown is 8.67E-08 /y. Details of the results are described in the next subsections.

25.8.6.2 Significant Contributors to Shutdown FDF

(1) Significant POSs

Figure 25.8.6-1 shows a summary contribution of POSs in the form of a pie chart. The Shutdown FDFs for the POSs are shown below.

POS S : 1.07E-09 /y

POS A : 9.09E-09 /y

POS B-1 : 4.88E-09 /y

POS B-2 : 1.46E-08 /y

POS C : 5.38E-08 /y

POS D : 3.21E-09 /y

POS C has the highest FDF (62 percent of the total Shutdown FDF). This POS is characterised by maintenance (out of service) of Class 1 Divisions 1 and 3 systems, closed RPV head, closed pool gate, and relatively low decay heat. The initiating event “Loss of Class 1 AC” in POS C contributes to 32 percent of the total FDF. In POS C, Divisions 1 and 3 systems are out of service. Only Division 2 systems are available.

The second contributor is POS B-2 (17 percent of the total Shutdown FDF). This POS is characterised by maintenance (out of service) of Class 1 Divisions 1 and 3 systems, open RPV head, and open pool gate.

The third contributor is POS A (11 percent of the total Shutdown FDF). This POS is characterised by maintenance (out of service) of Class 1 Division 2 systems, closed RPV head, closed pool gate, and relatively high decay heat.

The fourth contributor is POS B-1 (6 percent of the total Shutdown FDF). This POS is characterised by maintenance (out of service) of Class 1 Division 2 systems, open RPV head, open pool gate and the longest duration.

Despite the highest decay heat, POS S has the smallest Shutdown FDF due to the most redundant mitigation systems and the shortest duration (smallest POS weighting factor).

POS D has a higher FDF than POS S because the duration is 7.5 times longer than that of POS S, one division of the backup building systems is out of service, and a unique IE, RPV draindown during water level decreasing by CUW (HE), is considered.

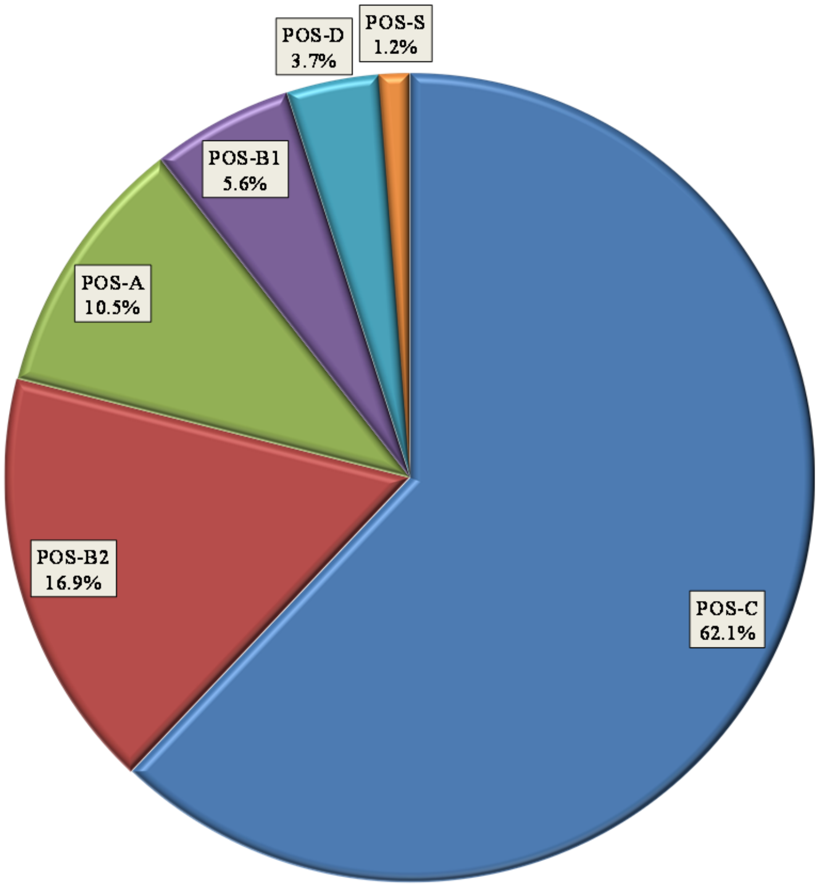

Figure 25.8.6-1 Contribution to Shutdown FDF by POS

(2) Significant Initiating Events

Table 25.8.6-1 shows the major contributors of initiating events to FDF.

The initiating events that have the largest contributors to FDF are identified to provide a perspective on the results.

Loss of Class 1 AC during POS C is the highest contributor: approximately 32 percent to the total Shutdown FDF. This IE disables RHR-B, HPCF-B, LPFL-B and MUWC-B, such that only FLSS and FLSR are available.

Loss of operating RHR frontline during POS C is the second contributor: approximately 17 percent to the total Shutdown FDF. This IE has the highest frequency among the IEs for POS C in the broad category "loss of decay heat removal".

LOCA (mechanical) above normal water level during POS B-2 is the third contributor: approximately 15 percent to the total Shutdown FDF. This IE only causes fuel damage at the SFP. The frequency of this IE is high and dominated by the human error probabilities regarding heavy load drop to the reactor well or S/D pit.

LOCA (mechanical) above normal water level during POS B-1 is the fourth contributor: approximately 4.5 percent to the total Shutdown FDF. This IE only causes fuel damage at the SFP. The frequency of this IE is high and dominated by the human error probabilities regarding heavy load drop to the reactor well or S/D pit.

Loss of R/BEEE/Z HVAC during POS C is the fifth contributor: approximately 4.2 percent to the total Shutdown FDF. The impact of this IE in the PSA model is identical to that of the "Loss of Class 1 AC during POS C" (the highest contributor). The difference of the contributions among these two IEs is caused by the difference of the IE frequencies.

A full listing and detailed description of the most significant IEs is provided in the Shutdown Level 1 PSA Topic Report [Ref-25.75].

Note: LOOP IEs (all POSs) contribute to only 5.1 percent of the total Shutdown FDF. This is because (1) the LOOP frequencies are generally smaller than the other IEs in the broad category "Loss of decay heat removal", (2) recovery of offsite power is considered, and (3) diverse on-site power (EDGs and BBGs) are considered.

IEs caused by human error(s) contribute to 23 percent of the total Shutdown FDF. This number includes the LOCA (mechanical) above normal water level, LOCA through MS line plug, and Loss of SFP inventory which are dominated by human error probabilities.

Table 25.8.6-1 Initiating Events Contributing more than 1 Percent to Shutdown FDF

Imitating Event	Broad Category	Contribution to FDF
Loss of AC Class 1E Bus Support System Initiating Event at POS C	Loss of Decay Heat Removal	32.1 %
Loss of RHR-B shutdown cooling mode Initiating Event at POS C	Loss of Decay Heat Removal	17.3 %
LOCA (mechanical) above normal water level	LOCA above Normal Water Level	14.9 %
LOCA (mechanical) above normal water level	LOCA above Normal Water Level	4.5 %
Loss of RBEEEZ HVAC (B) Support System Initiating Event at POS C	Loss of Decay Heat Removal	4.2 %
Loss of a Class 1 AC bus	Loss of Decay Heat Removal	2.2 %
Loss of RHR-A shutdown cooling mode Initiating Event at POS A	Loss of Decay Heat Removal	2.0 %
LOCA at feedwater line inside PCV at POS C	LOCA between normal water level and TAF	2.0 %
Loss of HNCW Initiating Event at POS A	Loss of Offsite Power	1.9 %
Loss of RHR-B shutdown cooling mode Initiating Event at POS D	Loss of Decay Heat Removal	1.8 %
LOCA at feedwater line inside PCV at POS A	LOCA between normal water level and TAF	1.6 %
Loss of RCW/RSW-A Support System Initiating Event at POS A	Loss of Decay Heat Removal	1.5 %
RPV draindown during water level decreasing by CUW	LOCA below TAF	1.5 %
Loss of HNCW Initiating Event at POS C	Loss of Offsite Power	1.4 %

(3) Significant Accident Sequences

Table 25.8.6-2 summarises the accident sequences which contribute more than 1 percent to the Shutdown FDF, the corresponding frequencies, and the percentage contribution to Shutdown FDF and accident class.

Figure 25.8.6-2 illustrates the FDF contribution in the form of a pie chart.

Those accident sequences which contribute more than 1 percent to the Shutdown FDF are described below.

1. LDHR-C07

LDHR-C07 is a low pressure, boil-off, S/P bypass sequence (Class: I).

- Operating RHR-B loses its function due to the IE.
- RHR-B is not able to be restarted due to dependent failure from the IE, frontline system failure, support system failure (including station blackout), and/or human error.
- The RPV is pressurised due to boiling, but the SRVs (safety valve function) successfully actuate and reseat and prevent a consequential LOCA.
- HPCF-B fails to provide high pressure injection.
- RPV depressurisation is performed manually or automatically (Transient ADS or RDCF).
- LPFL-B, MUWC-B, FLSS, and FLSR fail to provide low pressure injection.
- With no injection to the RPV, fuel uncovery and damage occur in the reactor.

This accident sequence contributes 39.8 percent of total Shutdown FDF.

2. LDHR-C09

LDHR-C09 is a consequential LOCA, boil-off, S/P bypass sequence (Class: I).

- Operating RHR-B loses its function due to the IE.
- RHR-B is not able to be restarted due to dependent failure from the IE, frontline system failure, support system failure (including station blackout), and/or human error.
- The RPV is pressurised due to boiling, and a consequential LOCA occurs because no SRV (safety valve function) opens.
- Given a consequential LOCA, no further mitigation is credited.
- The S/P is not available for fission product scrubbing because the containment boundary is not intact.

This accident sequence contributes 19.8 percent of total Shutdown FDF.

3. LOCA-ANWL-B204

LOCA-ANWL-B204 is a boil-off sequence with fuel damage in the SFP only (Class: VIII).

- The SFP water level begins to decrease due to a LOCA in the reactor side above NWL including a LOCA at the reactor well and S/D pit.
- Spill-out of the SFP water to the skimmer surge tank stops due to the decrease in the SFP water level, which results in loss of decay heat removal by the FPC system.
- The SFP water level further decreases to the elevation of the reactor well gate bottom, resulting in loss of SFP decay heat removal by RHR shutdown cooling mode due to decoupling of the reactor and SFP.
- Makeup to the SFP by FLSS, MUWC-B, and FLSR all fail.
- SPCU is not credited because AC-C is out of service.
- Fuel uncovery in the SFP results in fuel damage.

This accident sequence contributes 15.4 percent of total Shutdown FDF.

4. LDHR-A08

LDHR-A08 is a consequential LOCA, boil-off, S/P bypass sequence (Class: I). This sequence is similar to sequence LDHR-C09 except for the POS. This accident sequence contributes 4.7 percent of total Shutdown FDF.

5. LOCA-ANWL-B105

LOCA-ANWL-B105 is a boil-off sequence with fuel damage in SFP only (Class: VIII).

This sequence is similar to sequence LOCA-ANWL-B204 except for the POS and additional failure of SPCU. This accident sequence contributes 4.6 percent of total Shutdown FDF.

6. LDHR-A06

LDHR-A06 is a low pressure, boil-off, S/P bypass sequence (Class: I). This sequence is similar to sequence LDHR-C07 except for the POS. This accident sequence contributes 4.2 percent of total Shutdown FDF.

7. LDHR-D11

LDHR-D11 is a consequential LOCA, boil-off, S/P bypass sequence (Class: I). This sequence is similar to sequence LDHR-C09 except for the POS. This accident sequence contributes 3.1 percent of total Shutdown FDF.

8. LOCA-NWL-TAF-C07

LOCA-NWL-TAF-C07 is a low pressure, boil-off, S/P bypass sequence (Class: I).

- Operating RHR-B shutdown cooling mode is lost due to loss of suction or isolation signal.
- Automatic isolation of LOCA fails or LOCA is not isolatable.
- HPCF-B fails to provide high pressure injection.

- LPFL-B, MUWC-B, FLSR, and FLSS all fail.

This accident sequence contributes 2.0 percent of total Shutdown FDF.

9. LOCA-NWL-TAF-A05

LOCA-NWL-TAF-A05 is a low pressure, boil-off, S/P bypass sequence (Class: I). This sequence is similar to sequence LOCA-NWL-TAF-C07 except for the POS and no credit to FLSR. This accident sequence contributes 1.6 percent of total Shutdown FDF.

10. LDHR-B207

LDHR-B207 is a low pressure, boil-off, S/P bypass sequence with fuel damage in both the reactor and SFP (Class: VI).

- Operating RHR-B fails due to the IE.
- RHR-B is not able to be restarted due to dependent failure from the IE, frontline system failure, support system failure (including station blackout), and/or human error.
- Automatic makeup to the SFP skimmer surge tank by running the MUWC-B pump or by returning water from the skimmer surge tank to the SFP by running FPC-B fails.
- Injection to the RPV by HPCF-C fails.
- LPFL is not credited because the amount of water in the S/P may not be sufficient during pool gate open condition.
- Injection to the RPV or SFP by MUWC, FLSS, and FLSR all fail.
- There is no core cooling or SFP cooling or make-up, resulting in fuel damage in both the reactor and SFP.

This accident sequence contributes 1.2 percent of total Shutdown FDF.

Table 25.8.6-2 Fuel Damage Sequences Contributing more than 1 Percent to Shutdown FDF

Rank	Sequence	Class	Frequency (/y)	Contribution	Contribution to class
1	LDHR-C07	I	3.45E-08	39.8 %	51.6 %
2	LDHR-C09	I	1.72E-08	19.8 %	25.7 %
3	LOCA-ANWL-B204	VIII	1.33E-08	15.4 %	75.0 %
4	LDHR-A08	I	4.03E-09	4.7 %	6.0 %
5	LOCA-ANWL-B105	VIII	4.03E-09	4.6 %	22.6 %
6	LDHR-A06	I	3.66E-09	4.2 %	5.5 %
7	LDHR-D11	I	2.66E-09	3.1 %	4.0 %
8	LOCA-NWL-TAF-C07	I	1.73E-09	2.0 %	2.6 %
9	LOCA-NWL-TAF-A05	I	1.40E-09	1.6 %	2.1 %
10	LDHR-B207	VI	1.03E-09	1.2 %	75.0 %

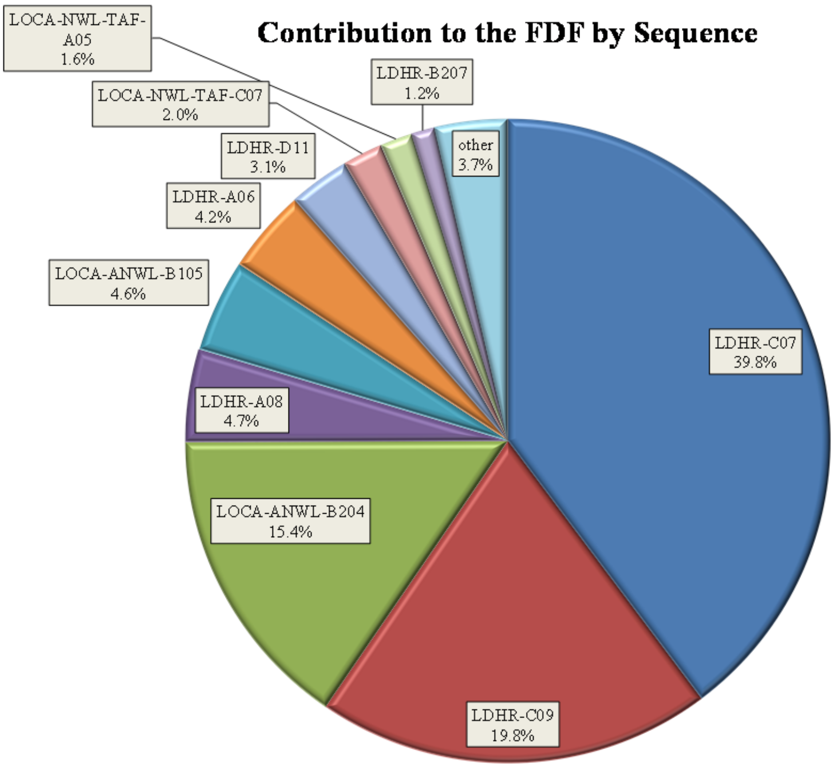

Figure 25.8.6-2 Contribution to Shutdown FDF by Top Ten Accident Sequence

(4) Significant Accident Classes

Table 25.8.6-3 shows the contribution of Accident Classes to Shutdown FDF. Figure 25.8.6-3 illustrates the FDF contribution in the form of a pie chart. Classes I to V are for POSs S, A, C and D where the pool gate is closed. Classes VI to IX are for POSs B-1 and B-2 where the pool gate is open (NOTE: the sequences of Class IX are truncated).

The highest contribution is Class I: Fuel damage due to boil-off at low pressure (with S/P bypass). This Class represents nearly 100 percent of the FDF for POSs S, A, C and D. This is because:

- Most of the fuel damage sequences are low pressure. Very limited sequence, i.e., success of RCPB protection followed by failure of RPV depressurisation, end with high pressure. This results in a small contribution of high pressure sequence (II).

- Most of the fuel damage sequences are boil-off sequences. Even in the LOCAs below TAF, the end states are assigned to Class I if onset of boiling is expected before fuel uncovery. This results in a small contribution of draindown sequence (III).

- S/P availability for fission product scrubbing is examined (credited) for only non-LOCA events during POSs S, D. This results in a small contribution of the S/P available sequences (IV, V).

Among the Classes for POSs B-1 and B-2, the highest contribution is Class VIII: Fuel damage in the SFP only due to boil-off. This is because the IEs where only SFP makeup is needed have high contribution, e.g., LOCA (mechanical) above normal water level.

Table 25.8.6-3 Shutdown FDF Contribution by Accident Class

Rank	Accident Class	Frequency (/y)	Contribution to FDF
1	I	6.67E-08	77.0 %
2	VIII	1.78E-08	20.5 %
3	VI	1.38E-09	1.6 %
4	VII	3.28E-10	0.4 %
5	IV	2.53E-10	0.3 %
6	II	1.54E-10	0.2 %
7	III	1.99E-11	0.0 %
8	V	8.67E-13	0.0 %

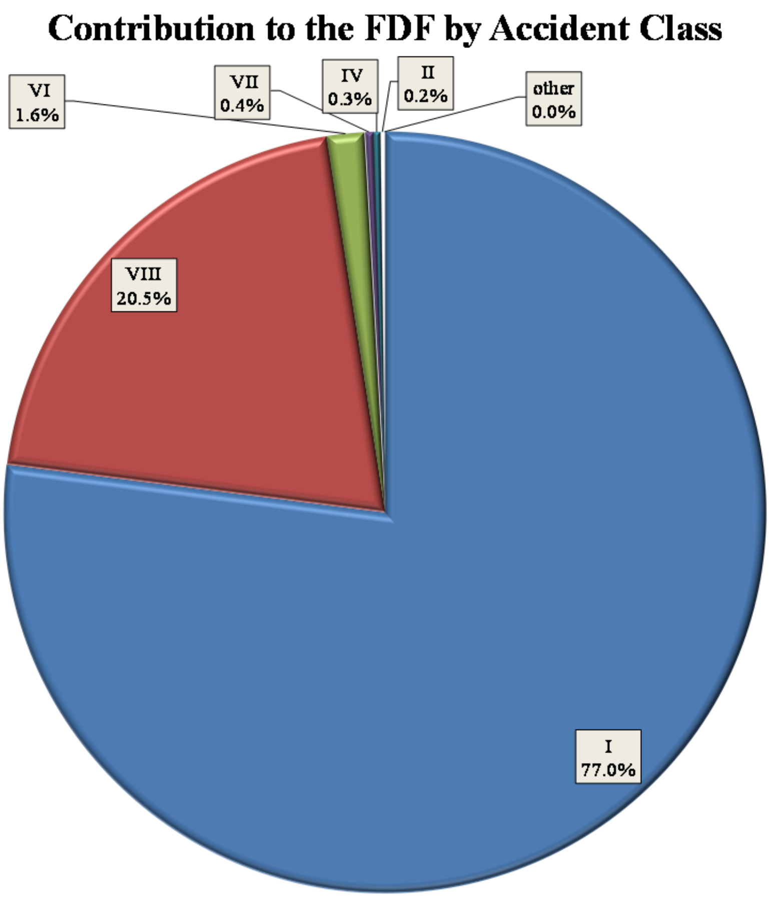

Figure 25.8.6-3 Contribution to Shutdown FDF by Accident Class

25.8.6.3 Importance Analysis

The following discussions provide the safety related importance of various systems, components and operator actions.

(1) Components important to safety (Fussell-Vesely Importance)

The components important to safety are discussed here and ranked according to F-V (a measure of how much a component's reliability directly influences the FDF). The components are listed in order of risk significance. A high F-V importance indicates that risk can be reduced by improving the component's reliability and/or availability. In this subsection, the initiating events, CCF events, test and maintenance events, human failure events and other marker events (e.g., IE markers, sequence markers, class markers, etc.) are excluded from the discussion. Table 25.8.6-4 shows the F-V importance of the top ten events.

RHR-B local cooling unit supply fan:

The highest F-V event is U41-FNS-RA-__-B104_S "Fan (Standby) B104_S Fail to Run > 1 Hour": the F-V is 0.13. This component is the supply fan B104 for the RHR-B pump room local cooling unit. Loss of this component causes loss of RHR-B. This basic event is the major contributor to the IE "Loss of operating RHR frontline during POS C" which contributes to 16 percent of the total Shutdown FDF.

Buses of M/C-1D, P/C-1D1 and P/C-1D2:

There are three components that have the second highest F-V value (0.092): R22-BSA-LF-__-M/C-1D "Bus (AC Power) M/C-1D Loss of Function", R23-BSA-LF-__-P/C-1D1 "Bus (AC Power) P/C-1D1 Loss of Function", R23-BSA-LF-__-P/C-1D2 "Bus (AC Power) P/C-1D2 Loss of Function". These basic events are the major contributors to the IE "Loss of Class 1 AC during POS C" which contributes to 28 percent of the total Shutdown FDF. The reason for the identical F-V value among these three basic events is the identical impact they have in the Shutdown PSA model.

NOTE: Loss of P/C-1D1 impacts HVAC-RBEEE-B/Z and thus causes loss of M/C-1D.

NOTE: Loss of P/C-1D2 impacts RSW-B pump and thus disables RHR-B, HPCF-B.

NOTE: Loss of P/C-1D2 also impacts MUWC-B which is powered from MCC under P/C-1D2.

Feedwater A line check valves:

There are two components that have the fifth highest F-V value (0.075): B21-CV_-CC-__-F051A "Check Valve F051A Fail to Open" and B21-CV_-CC-__-F052A "Check Valve F052A Fail to Open". These check valves are located on the feedwater A line. Failure to open each check valve disables RHR-A shutdown cooling mode, LPFL-A, FLSS and FLSR because these systems uses the feedwater A line. MUWC is also partially impacted because the RPV injection by MUWC is via the feedwater A line or the feedwater B line.

RHR-A return line check valve:

E11-CV_-CC-__-F006A "Check Valve F006A Fail to Open" has the identical impact to that of the above two components. This check vale is located on the RHR-A return line. This basic event is slightly lower F-V (0.072). This is because this check valve is included in CCF group of three check valves while the above two check valves form a CCF group.

FLSS RPV injection line check valves:

There are two components that have the eighth highest F-V value (0.072): E71-CV_-CC-__-F024 "Check Valve F024 Fail to Open" and E71-CV_-CC-__-F025 "Check Valve F025 Fail to Open". These check valves are located on the RPV injection line for FLSS (shared by two divisions). Failure to open each check valve disables RPV injection by both FLSS and FLSR.

Interface of SLU2-2:

The tenth highest F-V value (0.029) is H11-IF_-UF-__-L2C2 “Interface L2C2 Undetectable Loss of Function”. This event fails manual operation of RHR-B, MUWC-B and manual closure of the containment isolation valves controlled by the division II signal. Since POS C has the highest contributor, this event is higher importance than the corresponding Divisions I and III events.

Table 25.8.6-4 F-V Importance of Top Ten Components

Rank	Event	F-V	Description
1	U41-FNS-RA-__-B104_S	1.30E-01	Fan (Standby) B104_S Fail to Run > 1 Hour
2	R22-BSA-LF-__-M/C-1D	9.19E-02	Bus (AC Power) M/C-1D Loss of Function
2	R23-BSA-LF-__-P/C-1D1	9.19E-02	Bus (AC Power) P/C-1D1 Loss of Function
2	R23-BSA-LF-__-P/C-1D2	9.19E-02	Bus (AC Power) P/C-1D2 Loss of Function
5	B21-CV_-CC-__-F051A	7.53E-02	Check Valve F051A Fail to Open
5	B21-CV_-CC-__-F052A	7.53E-02	Check Valve F052A Fail to Open
7	E11-CV_-CC-__-F006A	7.23E-02	Check Valve F006A Fail to Open
8	E71-CV_-CC-__-F024	7.19E-02	Check Valve F024 Fail to Open
8	E71-CV_-CC-__-F025	7.19E-02	Check Valve F025 Fail to Open
10	H11-IF_-UF-__-L2C2	2.89E-02	Interface L2C2 Undetectable Loss of Function

(2) CCF events with F-V > 0.005

Rank	Event	F-V	Description
1	PR-OSRV_SD	2.84E-01	Failure of Pressure Relief by Opening Safety Relief Valves (for Shutdown PSA)
2	CCF-BBG	2.04E-02	CCF of loss of BBG
3	E71-FT-NH-FT004_ALL	5.14E-03	CCF of all components in group 'E71-FT-NH-FT004'

There are three significant CCF events.

The highest F-V event is PR-OSRV_SD “Failure of Pressure Relief by Opening Safety Relief Valves (for Shutdown PSA)” which is the failure to open all 16 SRVs. The F-V is 0.28. This is due to the conservative assumption on accident sequence modelling: Given a consequential LOCA, no further mitigation is credited [Ref-25.75], as well as CCF modelling: Only 8 SRVs are considered due to the limitation of available alpha factors for CCF.

Rank 2 and 3 events impact FLSS (System ID E71). This demonstrates that FLSS is important during an outage where redundancy of Class 1 systems is degraded during some POSs. The F-V values of CCF events are generally lower than those in the at Power PSA [Ref-25.9]. That is because system or component redundancy is degraded during an outage compared to at Power conditions.

The CCF by RAW is not presented here; it is dominated by partial combinations of component groups which would appear in cutsets along with a random independent failure affecting the remaining component(s) in the CCF group. These cutsets are significantly lower in frequency compared to similar cutsets with the total CCF of all components; if the CCF of the partial group is set to 1.0 (for RAW), then these cutsets have a significant increase in their frequency, which drives the high worth for RAW.

(3) Post-Initiator HFEs with F-V > 0.005

Rank	HFE	F-V	Description
1	___-HFE-SC-FL	1.80E-01	Failure of manual initiation of FLSS
2	FLSR-SD_ST	1.67E-01	FLSR (Mobile Injection Facility) Unavailability (Line-up before IE)
3	___-HFE-SC-MU	1.31E-01	Failure of manual initiation of MUWC
4	FLSR-SD	5.41E-02	FLSR (Mobile Injection Facility) Unavailability

Since FLSS is credited for all POSs and almost all IEs, its manual operation has a high F-V. Similar discussion is applicable to MUWC although the absolute values of importance measures are smaller than that of FLSS.

FLSR failure probability is represented by the associated human error probabilities (FLSR-SD, FLSR-SD_ST). It is assumed that FLSR is required to be in service (connected to the injection point) before starting POS B-2 and until completion of POS C [Ref-25.75]. This enables credit of FLSR in the accident sequences which do not have a time window of more than 8 hours for FLSR in other POSs, FLSR is credited for the accident sequences which have the time window for at least 8 hours for FLSR [Ref-25.75]. These HFEs have high importance values because they result in failure to provide loss of pressure injection make-up to the reactor core.

Manual initiation of RHR shutdown cooling mode is considered in limited IEs and accident sequences, the importance of this operator action is relatively smaller than the manual operation of FLSS and MUWC.

Since HPCF and LPFL cannot inject water into SFP, the importance values of manual operation for these systems (not listed above) are not as important as those of other systems which can inject into SFP.

(4) Pre-Initiator HFEs

There are no pre-initiator HFEs with F-V over 0.005.

(5) Initiating Event HFEs (TYPE B) with F-V > 0.005

Rank	HFE	F-V	Description
1	ANWL-M_B2	1.49E-01	LOCA (mechanical) above normal water level
2	ANWL-M_B1	4.50E-02	LOCA (mechanical) above normal water level
3	___-HFE-SB-AC_A	2.17E-02	Loss of a Class 1 AC bus
4	___-HFE-SB-CD_C	1.45E-02	RPV draindown during water level decreasing by CUW
5	___-HFE-SB-CD_D	5.31E-03	RPV draindown during water level decreasing by CUW

There are five initiating event HFEs with F-V over 0.005. The initiating events of which the frequency is dominated by human error probability are also included.

The rank 1 and rank 2 events for F-V are LOCA (mechanical) above normal water level during POS B-2 and POS B-1, respectively. Since the frequencies of these IEs are dominated by human error probabilities regarding heavy load drop, these IE basic events are included in the discussion of Type B HFEs. These IEs have relatively high F-V values because of the high IE frequency.

Similarly, LOCA through MS line plug IEs are dominated by associated human error probability "Leakage during MS/PCV inspection (HE)". Due to the smaller frequency and smaller leak rate, the importance values of these IEs are lower than those of LOCA (mechanical) above normal water level.

The rank 3 F-V event is the human error induced loss of AC but during POS A. This event has one order of magnitude higher frequency than that of the support system initiator “Loss of Class 1 AC” at POS A.

Rank 4, 5 F-V events are the human error induced RPV draindown events during POS C and D, respectively. The higher importance in POS C is mainly due to the offline condition of the Class 1 Division 1 and 3 systems.

All the maintenance-induced LOCAs below TAF, i.e., Leakage during FMCRD inspection, Leakage during RIP inspection, Leakage during replacement ICM nozzles do not match the definition of the significant basic event. Relatively small importance values compared to those of LOCA (mechanical) below TAF are owing to the realistic leak rate evaluation.

Since the frequencies of Loss of SFP inventory IEs are dominated by human error probabilities regarding heavy load drop, these IE basic events are included in the discussion of Type B HFEs. These IEs do not match the definition of significant Basie event. That is because Loss of SFP inventory is treated as design leakage ($30m^3/h$) such that automatic makeup function to the skimmer surge tank by MUWC is credited.

(6) System Level Importance

All the basic events that have the unique system ID (including pre-initiator HFEs) are assigned to one of the “systems” specified by the system ID. Some miscellaneous basic events, e.g., BBG-1, BBG-2, and FLSR unavailability are manually checked for specifying the relevant systems. The CCF events and post initiator HFEs are not included. The initiating event basic events are also not included. The sequence flag events, the accident class flag events, POS flag events are also not included. Note: Some minor systems, e.g., D11 “safety process radiation monitoring system” in which only a sensor is modelled as the source of spurious signal, are not shown.

System level importance values are summarised in Table 25.8.6-5 and Table 25.8.6-6.

The observations are as below.

F-V:

- SRV has the highest F-V because failure to open all the SRVs (safety valve function) prevents either letdown to the S/P and/or depressurisation of the RPV for low pressure injection.

- Power Centre (P/C) has the second highest F-V because the most dominant IE is Loss of AC power during POS C which contributes 32 percent to the total FDF.

- FLSR has the third highest F-V. FLSR has high unavailability and it is credited in all sequences with sufficient time margin.

- CONTROL PANEL (digital C&I components and software) has the fourth highest F-V. The reasons for the high F-V are wide use of digital C&I system for Class 1 and Class 3 safety systems and the conservative modelling of software CCF in terms of probability and consequence.

- FLSS has the fifth highest F-V. This is because FLSS is credited for all the sequences except for the most dominant sequences: RPV pressurisation and failure to open SRVs.

- Nuclear Boiler system (NB), excluding SRVs, has the sixth highest F-V. Various SSCs are involved: reactor vessel instrumentation, FW-A line check valves (forms injection line of RHR-A, FLSS, FLSR), etc.

- HVAC has the seventh highest F-V. The reason for the high F-V is that room cooling is needed to wide range of systems/components, e.g., HPCF, RHR, FLSS, EDG, BBG, RCW, RSW, TCW, TSW, buses, controllers.

- RHR has the eighth highest F-V. The system level F-V of RHR is smaller than the summed F-V of the initiating events "Loss of Shutdown Cooling Mode" because these IEs include the loss of RHR pump room local cooling units. The local cooling units belong to the HVAC system.

- Metal-Clad Switchgear (M/C) has the ninth highest F-V. The smaller F-V than that of P/C is due to the smaller number of components which belongs to M/C.

- EDG has the tenth highest F-V. The system level F-V is comparable to the total contribution from the IEs in the blood category "Loss of Offsite Power".

RAW:

- CUW has the seventh highest RAW. CUW is not credited as a mitigation system but causes initiating events, i.e., "LOCA at CUW outside PCV", "LOCA (mechanical) below TAF" and "LOCA and Spurious RPV draindown by CUW". Since the basic events contributing to these IEs are set to TRUE, the FDF significantly increases.

- MCC has the eighth highest RAW. Loss of MCC mainly impacts fans, motor operated valves and DC power supplies (through battery chargers).

- RCW has the tenth highest RAW. RCW supports RHR, HPCF and EDGs.

Table 25.8.6-5 System Level Importance (descending order for F-V)

Rank (F-V)	Systems	FV	RAW
1	SRV	2.86E-01	1.63E+00
2	PC	2.51E-01	2.01E+06
3	FLSR	2.21E-01	1.99E+01
4	FLSS	2.12E-01	1.40E+04
5	NB	2.01E-01	5.03E+04
6	HVAC	2.00E-01	7.66E+03
7	CONTROL PANELS(DIGITAL,RPS)	1.89E-01	5.06E+03
8	RHR	1.35E-01	1.38E+06
9	MC	1.25E-01	1.26E+04
10	DG	5.90E-02	1.85E+01
11	BBG	4.69E-02	1.41E+01
12	RCW	3.78E-02	1.49E+03
13	MCC	1.91E-02	5.56E+03
14	HECW	1.55E-02	9.23E+00
15	RSW	9.39E-03	3.03E+02
16	MUWC	6.61E-03	3.87E+00
17	HPCF	4.66E-03	4.87E+02
18	DC	3.60E-03	9.99E+01
19	TCW	1.45E-03	2.46E+01
20	HNCW	1.30E-03	2.71E+01
21	SPCU	1.27E-03	1.33E+00
22	TSW	1.10E-03	1.37E+01
23	EECW	1.01E-03	1.00E+00
24	CUW	7.59E-04	6.11E+03
25	ANT	7.35E-04	3.83E+01
26	FPC	5.96E-04	3.76E+00
27	SLC	5.13E-04	1.98E+00
28	AC	4.92E-07	1.00E+00

Table 25.8.6-6 System Level Importance (sorted by RAW)

Rank (RAW)	Systems	FV	RAW
1	PC	2.51E-01	2.01E+06
2	RHR	1.35E-01	1.38E+06
3	NB	2.01E-01	5.03E+04
4	FLSS	2.12E-01	1.40E+04
5	MC	1.25E-01	1.26E+04
6	HVAC	2.00E-01	7.66E+03
7	CUW	7.59E-04	6.11E+03
8	MCC	1.91E-02	5.56E+03
9	CONTROL PANELS(DIGITAL,RPS)	1.89E-01	5.06E+03
10	RCW	3.78E-02	1.49E+03
11	HPCF	4.66E-03	4.87E+02
12	RSW	9.39E-03	3.03E+02
13	DC	3.60E-03	9.99E+01
14	ANT	7.35E-04	3.83E+01
15	HNCW	1.30E-03	2.71E+01
16	TCW	1.45E-03	2.46E+01
17	FLSR	2.21E-01	1.99E+01
18	DG	5.90E-02	1.85E+01
19	BBG	4.69E-02	1.41E+01
20	TSW	1.10E-03	1.37E+01
21	HECW	1.55E-02	9.23E+00
22	MUWC	6.61E-03	3.87E+00
23	FPC	5.96E-04	3.76E+00
24	SLC	5.13E-04	1.98E+00
25	SRV	2.86E-01	1.63E+00
26	SPCU	1.27E-03	1.33E+00
27	AC	4.92E-07	1.00E+00
28	EECW	1.01E-03	1.00E+00

25.8.6.4 Level 1 PSA Results Insights

The quantification results provide the following results:

- POS C has the highest risk contribution because Class 1 divisions 1 and 3 systems are out of service. This is a natural insight clearly related to the assumed outage schedule defined in Section 25.8.2. Loss of AC power results in only FLSS and FLSR available for makeup. Modification of the outage schedule was recommended.

- POS B-2 is the second contributor because only Class 1 division 2 systems are, similar to POS C, automatic initiation of makeup system is not available, and unique IEs challenging SFP exists, e.g., LOCAs above NWL (dominated by heavy load drop), LOCAs at SFP (dominated by heavy load drop). It is recommended that the significant time margin for operator recovery actions in POS B-1 and B-2 should be adequately taken into account for the future improvement.

- Due to the partially degraded redundancy of the Class 1 systems, FLSS is important during outage as well as at Power. Capability of FLSS to SFP makeup also makes FLSS important during outage. The post-initiator HFE that has the highest F-V is the manual initiation of FLSS.

- FLSR also provides considerable risk reduction in outage. It is assumed FLSR is required to be connected to the injection point before an IE during POS B-2 and POS C per the recommendation by the experience system engineer and maintenance planners; such that redundancy of mitigation systems is improved under the condition Class 1 divisions 1 and 3 systems are in maintenance.

- Risk contribution by CCF events is generally lower than those in the at Power PSA [Ref-25.9]. That is because system or component redundancy is degraded during outage compared to at Power condition.

- Pre-initiator HFEs are not very risk significant, which is consistent with the at Power PSA.

- Initiating event HFEs (Type B) have relatively small risk contribution (up to about 2 percent of total Shutdown FDF), except for LOCA (mechanical) above normal water level which is dominated by heavy load drop frequency and has high frequency.

- Availability of RPV injection paths given overpressure failure of reactor coolant pressure boundary has large impact on the shutdown FDF risk. Sensitivity analysis was performed especially focusing on the dominant sequence: RPV pressurisation and failure to open all SRVs (safety valve function) (see Section 25.8.11.1).

25.8.7 Analysis for Shutdown Level 2 PSA

In the UK ABWR Shutdown Level 2 PSA, CETs have not been developed because the containment boundary is assumed always open throughout the outage period. Instead each PDS has been directly linked to one of the defined release categories.

As shown in Table 25.8.4-1, nine accident classes are defined, considering RPV head condition (open/close), containment head condition, pressure at fuel uncover (when RPV head is close), mechanism of fuel uncover (boil-off or draindown), place of fuel damage (reactor and/or SFP), and any other conditions regarding safety function considered. Since no sequence for the Class IX was quantified due to low frequency of the associated initiator (around 1E-17 /y), eight accident classes I to VIII are discussed as the starting point of the Level 1 to Level 2 interface.

Since the accident classes were defined by taking account of the Level 2 aspects, e.g., RPV head condition, S/P availability, the PDS for Level 2 PSA are almost identical to the accident classes. Expected release timing is additionally considered for the PDS I definition (PDS Ia: Late, PDS Ib: Early) in order to evaluate LERF. As shown in Table 25.8.7-1, the following PDSs are defined.

PDS Ia (It links accident class I with late fission product release)

PDS Ib (It links accident class I with early fission product release)

PDS II (It links to accident class II)

PDS III (It links to accident class III)

PDS IV(It links to accident class IV)

PDS V (It links to accident class V)

PDS VI (It links to accident class VI)

PDS VII (It links to accident class VII)

Note: The fuel damage sequences assigned to PDS VIII (fuel damage in SFP only at POS B-1 and B-2) are treated in the SFP Level 2 PSA in terms of both frequency and source term.

25.8.7.1 Mitigation Systems

RPV injection after fuel damage in reactor:

RPV injection is not credited for the Shutdown Level 2 PSA for the following reasons.

Since the containment boundary is assumed always open, occurrence of RPV breach (potentially prevented by RPV injection in adequate timing) does not impact the LRF or LERF.

RPV injection would have both positive (fission product scrubbing inside the RPV) and negative (increased steam release to the environment and hence increased fission product releases) influences on source term.

The severe accident analyses performed with/without the RPV injection under RPV/PCV heads close conditions indicated that case 1 (with RPV injection) had a higher source term. The RPV/PCV head open condition cases have only small differences in the source terms.

It is uncertain whether the source term is actually increased or decreased by RPV injection.

Due to the above limitation of the MAAP code, the source term cannot be reasonably distinguished with/without the RPV injection. It is judged that the source term is smaller when RPV injection is successful, and thus the MAAP analyses without the RPV injection are used for the bounding source term assessment.

Drywell spray:

D/W spray operation is not credited for the Shutdown Level 2 PSA.

Since the containment boundary is assumed always open, containment spray does not impact LRF or LERF.

RPV/PCV heads closed condition:

- For the conditions where fission product scrubbing by S/P is available, i.e., PDS IV and V, or Safety Relief Valve (SRV) tailpipe break at wetwell (W/W) airspace, fission products are not released through the D/W. Therefore, D/W spray does not impact the source term.

- For the conditions with containment bypass (e.g., BOC with isolation failure), the release path does not include the D/W. Therefore, D/W spray does not impact the source term.

- For the conditions with an RPV bottom LOCA (including RPV rupture), the release path does not include the upper D/W, i.e., RPV -> lower D/W -> Access tunnel hatch -> RB. Therefore, D/W spray does not impact the source term.

- For the conditions with LOCAs above TAF or SRV tailpipe break at D/W, fission product release paths includes upper D/W and thus D/W spray may reduce the release amount. However, this effect is expected to be limited (not all aerosol removed).

RPV/PCV heads open condition:

- The major release path is from the RPV directly to the refuelling floor (not through D/W). Therefore, D/W spray does not impact the source term.

Lower Drywell Flooder System (LDF):

LDF operation is not credited for all the POSs and PDSs related to shutdown.

Containment venting:

Since the containment boundary is assumed always open for the shutdown cases, containment venting does not impact the LRF or LERF. The source term would also not be impacted due to the existing large openings.

Hydrogen management

Flammable gas (hydrogen) management measures are provided for the secondary containment. The flammable gas control strategy for the reactor building (secondary containment) in the UK ABWR consists of:

- Small amount of hydrogen leakage to the R/B : SGTS,

- Large amount of hydrogen leakage to the R/B : R/B Opening with defense shield, and

- Further reduction of hydrogen concentration to the R/B : PAR.

SGTS:

SGTS operation is not credited for the Shutdown Level 2 PSA. SGTS can treat a limited volume of hydrogen. The amount of hydrogen generation is large under the accident conditions considered in Shutdown PSA. Therefore, it is not credited for the Shutdown Level 2 PSA.

Passive Auto-catalytic Recombiner (PAR):

A PAR is a device that recombines the hydrogen gas and oxygen gas without a need for external power or operator action. The PARs are not expected to be effective for the accident condition in the Shutdown

Level 2 PSA, i.e., large amount of hydrogen released to the R/B though the open containment boundary. Therefore, it is conservatively not credited for the Shutdown Level 2 PSA.

Blow out panel:

A blowout panel is located in the refuelling floor (operating deck). It is opened when the operation deck pressure differential exceeds 3.43 [kPa] [gauge] or by operator manual action.

The blowout panel is assumed to open with the probability of 1.0 in the Shutdown Level 2 PSA for the following reasons:

- Decay heat and resulting steam release rate in the reactor side is much higher than that in the SFP side, such that it is more likely the blowout panel opens given the fuel damage in the reactor side.
- Hydrogen generation given fuel damage is much larger in the reactor side than that in SFP side, such that the blowout panel status would not impact the occurrence of hydrogen combustion. NOTE: Hydrogen combustion in the containment or R/B is deterministically modelled in the source term analysis.
- If the blowout panel is open, the release from the R/B to the environment is enhanced. Therefore, assuming the open condition from the beginning of the accident results in the bounding source term [Ref-25.77].

25.8.7.2 Operator Actions

Since available mitigation systems are already credited in the Shutdown Level 1 PSA, additional operator actions are not credited.

25.8.7.3 Severe Accident Phenomena

The reasons for not probabilistically evaluating the branch probabilities for the severe accident phenomena for the Shutdown Level 2 PSA are discussed here. Also, the interaction of these phenomena and source term is discussed.

Overpressure and Overtemperature:

The PCV is not pressurised during shutdown because of the open status from the beginning of the accident progression (including Level 1 analysis).

Molten Core Concrete Interaction (MCCI):

Since the containment boundary is assumed always open, MCCI does not impact LRF or LERF. Detailed increases in fission product releases were considered in the source term analysis by the MAAP code consideration of MCCI.

In-vessel Fuel Coolant Interaction (FCI):

Since the containment boundary is assumed always open, in-vessel FCI does not impact LRF or LERF.

For POSs A, C, B-1 and B-2 where the RPV and PCV heads are open, occurrence of in-vessel FCI would not impact the release path and source term because there is already a large release path from the RPV to refuelling floor.

For POSs S and D with S/P bypass conditions, e.g., LOCAs, BOCs, isolation failure, SRV tailpipe failure, there is already release path(s) from RPV to environment (not through the S/P). Although occurrence of in-vessel FCI may generate an additional release path, i.e., RPV directly to refuelling floor, this release path is already considered in the bounding source term analysis. For the IE “RPV rupture during hydrostatic test”, which is considered as PDS III, in-vessel FCI is not expected due to the rapid loss of all RPV water before fuel heatup.

For POSs S and D with the S/P available condition (PDS IV and V), occurrence of in-vessel FCI may generate an additional release path, i.e., RPV directly to refuelling floor, which may increase the release amount because S/P bypass occurs. Therefore, in-vessel FCI should be probabilistically addressed for PDSs IV and V in order to distinguish the release categories with available S/P and direct release to the refuelling floor. The same probability as used for the IEAP Level 2 PSA [Ref-25.10] is applied: 1.0E-04 for the following reasons.

- The in-core geometry (including fuel) is the same.
- The lower plenum coolant is saturated at the moment of fuel relocation for both at Power and shutdown conditions.

Ex-vessel Fuel Coolant Interaction (FCI):

Since the containment boundary is assumed always open, ex-vessel FCI does not impact LRF or LERF.

For all the cases ex-vessel FCI would not impact the release path and source term because there is already a release path from the RPV or PCV to the R/B at the moment of RPV breach.

Even if fission product scrubbing by the S/P is initially available (PDS IV and V), this effect is not significant after the RPV breach due to the more direct release path available.

DCH:

Since the containment boundary is assumed always open, DCH does not impact LRF or LERF.

For all the cases DCH would not impact the release path and source term because there is already a release path from the RPV or PCV to the R/B at the moment of RPV breach.

Even if fission product scrubbing by the S/P is initially available (PDSs IV and V), this effect is not effective after the RPV breach due to the more direct release path available.

Hydrogen burning and explosions:

Since the containment boundary is assumed to be always open, hydrogen burning at containment or the refuelling floor does not impact LRF or LERF.

SGTS and PAR operation were not credited for the Shutdown Level 2 PSA [Ref-25.77]. The blowout panel is assumed to be always open for deriving the bounding release amount [Ref-25.77]. Occurrence of hydrogen combustion in the containment or refuelling floor is deterministically judged and considered for the source term analysis by the MAAP code [Ref-25.77].

Recriticality during in-vessel recovery:

The potential for the boron carbide in the control blades to form a eutectic with steel at 1,500 K and relocate has been indicated from experiments. This is much less than the temperature at which the fuel starts to melt (2,500 K). Therefore, as the core heats up and begins to melt, there may be regions of the core which could be uncontrolled. If water was injected after the onset of control blade relocation, there is a potential for regions of the core to become critical and increase heat generation above decay heat levels. The probability of recovering core cooling in this interval is judged negligible, and further evaluation is not performed in this study.

25.8.7.4 Release Category

Based on the PDS and modelling considerations [Ref-25.77], the release categories are defined to cover all the fuel damage (and release) sequences and adequately distinguish the source terms.

Release Category for PDS Ia:

PDS Ia includes several release paths [Ref-25.77]. The RPV/PCV head open condition, e.g., POSs A and C, are dominant in terms of frequency. This condition should also give the bounding release amount and timing due to the largest opening to the R/B.

Decay heat for the beginning of POS D is commonly applied for all the MAAP analyses [Ref-25.77]. Therefore, differences in the decay heat level among the POSs do not reflect the release timing.

One release category is defined for this PDS: BO-SDO-L (Late release at RPV open or S/P bypass condition by fuel damage due to boil-off).

Release Category for PDS Ib:

PDS Ib includes several release paths [Ref-25.77]. The RPV/PCV head open conditions, e.g., POSs A and C, are dominant in terms of frequency. This condition should also give the bounding release amount and timing due to the largest opening to the R/B.

Decay heat for the beginning of POS D is commonly applied for all the MAAP analyses [Ref-25.77].

One release category is defined for this PDS: BO-SDO-E (Early release at RPV open or S/P bypass condition by fuel damage due to boil-off).

Release Category for PDS III:

PDS III includes one fuel damage sequence. Since the timing of fuel damage and release is much earlier than for PDSs Ia, Ib and II, the separate release category is defined: DD-SDO (Early release at S/P bypass condition by fuel damage due to draindown).

Release Category for PDS IV:

PDS IV includes two similar sequences [Ref-25.77]. Both have the same release path: RPV→ SRV→ S/P→ W/W→ RB (through hatches etc.)→ Environment.

Since the fission product scrubbing by the S/P has an impact on the release amount, the release category for this PDS is separately defined from those for PDSs Ia, Ib, II and III. The release category is BO-SDL (late release by fuel damage due to boil-off at low pressure with S/P available).

If in-vessel FCI occurs, the direct release path from the RPV to the refuelling floor occurs. Therefore, the same release category for PDS Ia is applied: BO-SDO-L.

Release Category for PDS V:

PDS V includes one sequence [Ref-25.77]. One release path is considered: RPV→ SRV→ S/P→ W/W→ RB (through hatches etc.)→ Environment.

Since the RPV pressure has an impact on the release amount [Ref-25.77], the release category for this PDS is separately defined from that for PDS IV. The release category is BO-SDH (Late release by fuel damage due to boil-off at high pressure with S/P available).

If in-vessel FCI occurs [Ref-25.77], the direct release path from the RPV to refuelling floor occurs. Therefore, the same release category for PDS Ia is applied: BO-SDO-L.

Release Category for PDS VI:

Reactor: PDS VI includes 18 sequences [Ref-25.77]. Although there are several release paths, all of the sequences have the common path of the RPV directly to the refuelling floor, and this release path should dominate the amount / timing of release due to having the biggest opening to the R/B.

The release path for PDS VI is similar to the bounding one for PDS Ia because of the similarities: RPV head open and fuel damage by boil-off. Therefore the release category BO-SDO-L is applied.

Release Category for PDS VII:

PDS VII is similar to PDS Ia in terms of fuel damage mechanism (boil-off), fuel damage timing (late), and major release path (RPV directly to refuelling floor). Therefore, the same release category BO-SDO-L is applied.

The linkage among the accident classes, PDSs, release categories and source term analysis cases and source term results are summarised in Table 25.8.7-1 and Table 25.8.7-2. The total frequency for each release category is listed in Table 25.8.7-3.

Table 25.8.7-1 Linkage between Accident Classes, PDSs, Release Categories and Source Term Analysis Cases

Accident Class	POS	PDS	Release Category	Inclusion in LER
I	S,A,C,D	Ia	BO-SDO-L	No
		Ib	BO-SDO-E	Yes
II	A,C	Ia	BO-SDO-L	No
	C	Ib	BO-SDO-E	Yes
	S,D	II	BO-SDO-L	No
III	C	III	DD-SDO	Yes
IV	S,D	IV	w/o in-vessel FCI: BO-SDL	No
			w/ in-vessel FCI: BO-SDO-L	No
V	D	V	w/o in-vessel FCI: BO-SDH	No
			w/ in-vessel FCI: BO-SDO-L	No
VI	B-1,B-2	VI	Reactor: BO-SDO-L	No
			SFP: BO-SFP_LOCA 02	No
VII	B-1,B-2	VII	Reactor: BO-SDO-L	No

Table 25.8.7-2 Source Term of Release Categories in Shutdown Mode

RC	Case No.	Time to Fuel Damage [hr]	Time to Core Support Failure [hr]	Time to RPV Failure [hr]	Time to Hydrogen Burn [hr]	LER	FP Mass Fraction Released to Environment											
							Noble Gas	CsI	TeO_2	SrO	MoO_2	CsOH	BaO	La_2O_3	CeO_2	Sb	Te_2	UO_2
BO-SDH	Case 3	34.1 hr	43.3 hr	44.6 hr	34.8 hr	No	1.00E+00	6.88E-02	9.87E-03	1.22E-02	9.66E-06	1.14E-02	5.37E-03	2.02E-04	5.03E-03	1.13E-01	5.96E-03	1.61E-05
BO-SDL	Case 3-3	34.1 hr	42.5 hr	48.1 hr	34.8 hr	No	1.00E+00	9.49E-02	3.11E-02	2.72E-02	2.45E-05	4.52E-02	1.17E-02	8.31E-04	1.25E-02	1.83E-01	6.42E-03	5.84E-05
DD-SDO	Case 3-5	24.3 hr	29.6 hr	37.3 hr	37.3 hr	Yes	1.00E+00	8.83E-01	3.37E-01	2.45E-02	2.28E-02	5.25E-01	3.07E-02	4.65E-04	4.31E-03	8.18E-01	1.17E-02	1.47E-05
BO-SDO-E	Case 6-1	26.3 hr	31.9 hr	37.3 hr	37.3 hr	Yes	1.00E+00	9.01E-01	4.58E-01	2.78E-02	2.65E-02	3.96E-01	3.89E-02	5.13E-04	4.73E-03	8.18E-01	1.52E-02	1.62E-05
BO-SDO-L	Case 6-2	28.8 hr	36.7 hr	45.9 hr	-	No	1.00E+00	8.11E-01	5.99E-01	1.06E-01	1.57E-01	7.03E-01	9.80E-02	3.95E-03	3.44E-02	6.35E-01	3.91E-03	2.12E-04
BO-SFP_LOCA 02 (SFP)	Case 1-1 (SFP)	406 hr	-	-	-	No	9.70E-01	9.41E-01	8.94E-01	7.29E-01	7.90E-01	9.41E-01	7.54E-01	4.13E-01	4.43E-01	8.66E-01	0.00E+00	0.00E+00

The frequency of BO-SFP_LOCA 02 is included in that of BO-SDO-L.

Table 25.8.7-3 Summary of Release Category Frequencies

Release Category	Frequency (/y)	Inclusion in LER
BO-SDO-L	6.69E-08	No
BO-SDO-E	1.72E-09	Yes
DD-SDO	2.00E-11	Yes
BO-SDL	2.53E-10	No
BO-SDH	1.03E-12	No
BO-SFP_LOCA 02	1.38E-09	No

25.8.8 Results (Shutdown Level 2)

This section summarises the quantification of Internal Events Shutdown Level 2 PSA for the UK ABWR. The contents present the quantification results including the LRF, LERF, dominant POSs, initiators, accident sequences, and basic event importance measures.

25.8.8.1 Model Results Summary

As a result of quantification, the total LRF is 6.89E-08 /y when a truncation value of 1.0E-14 /y was used. The total LERF (for release categories BO-SDO-E and DD-SDO) is 1.74E-09 /y.

The LRF is smaller than the FDF (8.67E-08). The difference is due to the treatment of Class/PDS VIII: included in the Shutdown Level 1 PSA but not included in the Shutdown Level 2 PSA.

Since no additional operator action is credited for the Shutdown Level 2 PSA, the human dependencies that have been investigated in the Shutdown Level 1 PSA [Ref-25.75] are also applicable to the Shutdown Level 2 PSA.

25.8.8.2 Significant Contributors to LRF

(1) Significant POSs

Figure 25.8.8-1 shows a summary contribution of POSs in the form of a pie chart. LRFs for POSs are shown below.

POS S : 1.07E-09 /y

POS A : 9.09E-09 /y

POS B-1 : 5.55E-10 /y

POS B-2 : 1.15E-09 /y

POS C : 5.38E-08 /y

POS D : 3.21E-09 /y

POS C has the highest LRF (78 percent of the total LRF). This POS is characterised by maintenance (out of service) of Class 1 Divisions 1 and 3 systems, closed RPV head, closed pool gate, and relatively low decay heat. The initiating event "Loss of Class 1 AC" in POS C contributes to 40 percent of the total LRF. In POS C, Divisions 1 and 3 systems are out of service. Only Division 2 systems are available.

The second contributor is POS A (13 percent of the total LRF). This POS is characterised by maintenance (out of service) of Class 1 Division 2 systems, closed RPV head, closed pool gate, and relatively high decay heat.

The third contributor is POS D (4.7 percent of the total LRF). POS D has a higher LRF than POS S because the duration is 7.5 times longer than that of POS S, one division of the backup building systems is out of service, and a unique IE, RPV draindown during water level decreasing by CUW (HE), is considered.

Since the Class VIII is moved from the Shutdown Level 2 PSA to the SFP Level 2 PSA [Ref-25.76], the contribution from POS B-2 is reduced to 1.7 percent of the total LRF while POS B-2 (including Class VIII) is the second contributor to the total FDF [Ref-25.75]. This POS is characterised by maintenance (out of service) of Class 1 Divisions 1 and 3 systems, closed RPV head, and open pool gate.

Despite the highest decay heat, POS S has the second smallest LRF due to the most redundant mitigation systems and the shortest duration (smallest POS weighting factor).

Since the Class VIII is moved from the Shutdown Level 2 PSA to the SFP Level 2 PSA [Ref-25.76], POS B-1 is the smallest contributor (0.8 percent of the total LRF) while this POS is the fourth contributor to the total FDF.

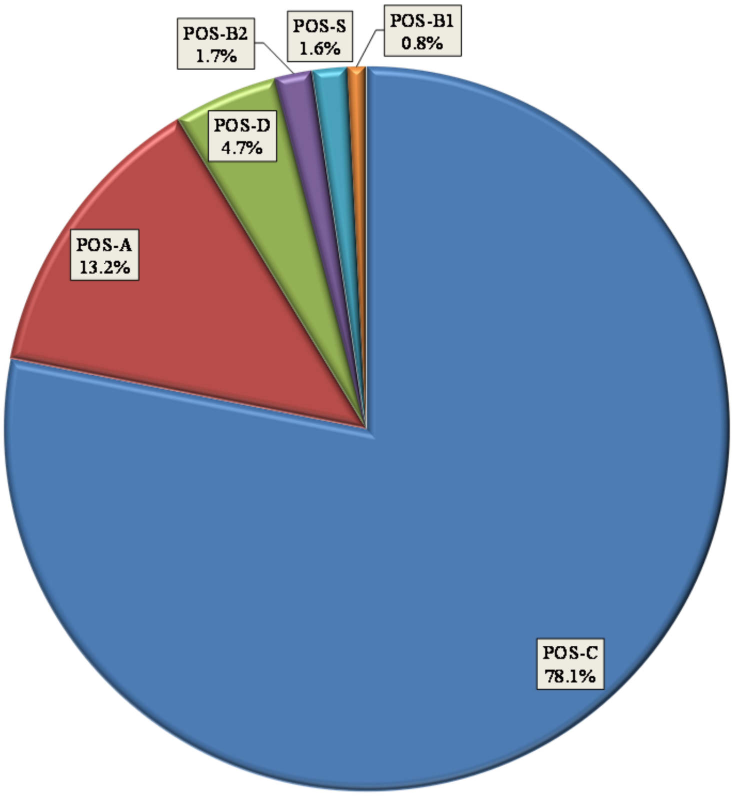

Figure 25.8.8-1 Contribution to LRF by POS

(2) Significant Initiating Events

Table 25.8.8-1 shows the initiating events contributing more than 1 percent to LRF.

Loss of Class 1 AC during POS C is the highest contributor: approximately 40 percent to the total LRF. This IE disables Residual Heat Removal System (RHR)-B, High Pressure Core Flooder System (HPCF)-B, Low Pressure Flooder System (LPFL)-B and Make-up Water Condensate System (MUWC)-B pump, such that only FLSS and FLSR are available.

Loss of operating RHR frontline during POS C is the second contributor: approximately 22 percent to the total LRF. This IE has the highest frequency among the IEs for POS C in the broad category "loss of decay heat removal".

Loss of R/BEEE/Z HVAC during POS C is the third contributor: approximately 5.3 percent to the total LRF. The impact of this IE in the PSA model is identical to that of the "Loss of Class 1 AC during POS C". The difference of the contributions among these two IEs is caused by the difference of the IE frequencies.

Loss of Class 1 AC bus (HFE) during POS A is the fourth contributor: approximately 2.7 percent to the total LRF. The impact of this IE in the PSA model is loss of AC power to the Division 1 systems, failing RHR-A, MUWC-A, C and SPCU.

Loss of operating RHR frontline during POS A is the fifth contributor: approximately 2.5 percent to the total LRF. This IE has the highest frequency among the IEs for POS A in the broad category "loss of decay heat removal".

LOCA at the feedwater line inside the PCV during POS C is the sixth contributor: approximately 2.5 percent to the total LRF. This IE disables RHR-B due to possible isolation signals (although the elevation of the feedwater line is above the RHR suction line), and also disables FLSS and FLSR due to loss of the injection path. Only HPCF-B, LPFL-B and MUWC-B are credited.

Loss of HVAC Normal Cooling Water System (HNCW) during POS A is the seventh contributor: approximately 2.4 percent to the total LRF. This IE causes a LOOP. The frequency of this IE is higher than those of generic LOOP events, and recovery of offsite power is not considered in this IE.

Loss of operating RHR frontline during POS D is the eighth contributor: approximately 2.3 percent to the total LRF. This IE has the highest frequency among the IEs for POS D in the broad category "loss of decay heat removal".

LOCA at the feedwater line inside the PCV during POS A is the ninth contributor: approximately 2.0 percent to the total LRF. This IE disables RHR-A and FLSS due to loss of injection line. RHR-C shutdown cooling mode is also disabled due to possible isolation signals (although the elevation of the feedwater line is above the RHR suction line). Only HPCF-C, LPFL-C and MUWC are credited.

Loss of RCW/RSW-A Support System Initiating Event at POS A is the tenth contributor. This contributes approximately 1.8 percent to the total LRF. Since this event does not fail the MUWC pumps, the impact of this event is smaller than that of the fourth contributor: Loss of Class 1 AC bus (HFE) during POS A.

Generic LOOP IEs (4 categories times 6 POSs) contribute to only 3.8 percent of the total LRF. This is because (1) the LOOP frequencies are generally smaller than the IEs in the broad category "Loss of decay heat removal", (2) recovery of offsite power is considered, and (3) EDGs and BBGs are considered.

IEs caused by human error(s) contribute to 5.3 percent of the total LRF. This number includes leakages during several kinds of inspections such as CUW, Emergency Core Cooling System (ECCS)/RHR, FMCRD and Reactor Internal Pump (RIP), leakage during replacement In-Core Monitor (ICM) nozzles, leakage during / failure of switchover of RHR shutdown cooling mode, loss of Class 1 AC bus, and RPV draindown during water level decreasing by CUW.

Table 25.8.8-1 Initiating Events Contributing more than 1 Percent to Shutdown LRF

Imitating Event	Contribution to LRF
Loss of AC Class 1E Bus Support System Initiating Event at POS C	40.4 %
Loss of RHR-B shutdown cooling mode Initiating Event at POS C	21.8 %
Loss of RBEEEZ HVAC (B) Support System Initiating Event at POS C	5.3 %
Loss of a Class 1 AC bus	2.7 %
Loss of RHR-A shutdown cooling mode Initiating Event at POS A	2.5 %
LOCA at feedwater line inside PCV at POS C	2.5 %
Loss of HNCW Initiating Event at POS A	2.4 %
Loss of RHR-B shutdown cooling mode Initiating Event at POS D	2.3 %
LOCA at feedwater line inside PCV at POS A	2.0 %
Loss of RCW/RSW-A Support System Initiating Event at POS A	1.8 %
RPV draindown during water level decreasing by CUW	1.8 %
Loss of HNCW Initiating Event at POS C	1.7 %
Loss of RCW/RSW-B Support System Initiating Event at POS C	1.1 %

(3) Significant Accident Sequences

Adding three Level 1 sequences for PDSs IV or V with in-vessel FCI (conditional probability of 1.0E-04): LDHR-D07-FCI, LDHR-S05-FCI and LDHR-D09-FCI, there are 46 large release sequences in total.

Table 25.8.8-2 summarises the accident sequences contributing more than 1 percent to the Shutdown LRF.

Figure 25.8.8-2 illustrates the LRF contribution in the form of a pie chart. The significant 9 accident sequences are described below.

1. LDHR-C07

LDHR-C07 is a low pressure, boil-off, S/P bypass sequence (PDS: Ia, RC: BO-SDO-L).

- Operating RHR-B loses its function due to the IE.
- RHR-B is not able to be restarted due to dependent failure from the IE, frontline system failure, support system failure (including station blackout), and/or human error.
- The RPV is pressurised due to boiling, but the SRVs (safety valve function) successfully actuate and reseat and prevent a consequential LOCA.
- HPCF-B fails to provide high pressure injection.
- RPV depressurisation is performed manually or automatically (Transient ADS or RDCF).

- LPFL-B, MUWC-B, FLSS, and FLSR fail to provide low pressure injection.
- With no injection to the RPV, fuel uncovery and damage occur in the reactor.

The main release pathway to the environment is the RPV, refuelling floor and blowout panel.

This accident sequence contributes 50 percent to the total LRF.

2. LDHR-C09

LDHR-C09 is a consequential LOCA, boil-off, S/P bypass sequence (PDS: Ia, RC: BO-SDO-L).

- Operating RHR-B loses its function due to the IE.
- RHR-B is not able to be restarted due to dependent failure from the IE, frontline system failure, support system failure (including station blackout), and/or human error.
- The RPV is pressurised due to boiling, and a consequential LOCA occurs because no SRV (safety valve function) opens.
- Given a consequential LOCA, no further mitigation is credited.
- The S/P is not available for fission product scrubbing because the containment boundary is not intact.
- The main release pathway to the environment is the RPV, refuelling floor and blowout panel.

This accident sequence contributes 25 percent to the total LRF.

3. LDHR-A08

LDHR-A08 is a consequential LOCA, boil-off, S/P bypass sequence (PDS: Ia, RC: BO-SDO-L). This sequence is similar to sequence LDHR-C09. This accident sequence contributes 5.9 percent to the total LRF.

4. LDHR-A06

LDHR-A06 is a low pressure, boil-off, S/P bypass sequence (PDS: Ia, RC: BO-SDO-L). This sequence is similar to sequence LDHR-C07 except for the POS. This accident sequence contributes 5.3 percent to the total LRF.

5. LDHR-D11

LDHR-D11 is a consequential LOCA, boil-off, S/P bypass sequence (PDS: Ia, RC: BO-SDO-L). This sequence is similar to sequence LDHR-C09 except for the POS. This accident sequence contributes 3.9 percent to the total LRF.

6. LOCA-NWL-TAF-C07

LOCA-NWL-TAF-C07 is a low pressure, boil-off, S/P bypass sequence (PDS: Ia, RC: BO-SDO-L).

- Operating RHR-B shutdown cooling mode is lost due to loss of suction or isolation signal.

- Automatic isolation of the LOCA fails or LOCA is not isolable.
- HPCF-B fails to provide high pressure injection.
- LPFL-B, MUWC-B, FLSR, and FLSS all fail.

The main release pathway to the environment is the RPV, refuelling floor and blowout panel.

This accident sequence contributes 2.5 percent to the total LRF.

7. LOCA-NWL-TAF-A05

LOCA-NWL-TAF-A05 is a low pressure, boil-off, S/P bypass sequence (PDS: Ib, RC: BO-SDO-E). This sequence is similar to sequence LOCA-NWL-TAF-C07 except for the POS and no credit for FLSR. This accident sequence contributes 2.0 percent to the total LRF.

8. LDHR-B207

LDHR-B207 is a low pressure, boil-off, S/P bypass sequence with fuel damage in both the reactor and SFP (PDS: VI, RC: BO-SDO-L for reactor and BO-SFP_LOCA 02).

- Operating RHR-B fails due to the IE.
- RHR-B is not able to be restarted due to dependent failure from the IE, frontline system failure, support system failure (including station blackout), and/or human error.
- Automatic makeup to the SFP skimmer surge tank by the running MUWC-B pump or return of water from the skimmer surge tank to SFP by the running FPC-B fails.
- Injection to the RPV by HPCF-C fails.
- LPFL is not credited because the amount of water in the S/P may not be sufficient during the pool gate open condition.
- Injection to the RPV or SFP by MUWC, FLSS, and FLSR all fail.
- There is no core cooling or SFP cooling or make-up, resulting in fuel damage in both the reactor and SFP.
- The main release pathway from the reactor to the environment is the RPV, refuelling floor and blowout panel.
- The main release pathway from the SFP to the environment is the refuelling floor and blowout panel.

This accident sequence contributes 1.5 percent to the total LRF.

9. LDHR-S09

LDHR-S09 is a consequential LOCA, boil-off, S/P bypass sequence (PDS: Ia, RC: BO-SDO-L). This sequence is similar to sequence LDHR-C09. This accident sequence contributes 1.1 percent to the total LRF.

Table 25.8.8-2 Large Release Sequences Contributing more than 1 percent to Shutdown LRF

L1 PSA end state	Frequency (/y)	Contribution to LRF	Expected release pathway	Earliest timing of uncovery from IE (h)	Release timing	Class	PDS	RC
LDHR-C07	3.45E-08	50.0 %	RPV→ Refuelling floor→ Env.	13.6	Late	I	Ia	BO-SDO-L
LDHR-C09	1.72E-08	24.9 %	RPV→ Refuelling floor→ Env. RPV→ D/W (cons. LOCA) → RB (through hatches etc.)→ Env.	13.6	Late	I	Ia	BO-SDO-L
LDHR-A08	4.03E-09	5.9 %	RPV→ Refuelling floor-> Env. RPV→ D/W (cons. LOCA)-> RB (through hatches etc.)→ Env.	5.2	Late	I	Ia	BO-SDO-L
LDHR-A06	3.66E-09	5.3 %	RPV→ Refuelling floor-> Env.	5.2	Late	I	Ia	BO-SDO-L
LDHR-D11	2.66E-09	3.9 %	RPV→ D/W (cons. LOCA) → RB (through hatches etc.)→ Env.	14.8	Late	I	Ia	BO-SDO-L
LOCA-NWL-TAF-C07	1.73E-09	2.5 %	RPV→ Refuelling floor→ Env. RPV→ D/W (LOCA) → RB (through hatches etc.)→ Env.	5	Late	I	Ia	BO-SDO-L
LOCA-NWL-TAF-A05	1.40E-09	2.0 %	RPV→ Refuelling floor→ Env. RPV→ D/W (LOCA) → RB (through hatches etc.)→ Env.	1.91	Early	I	Ib	BO-SDO-E
LDHR-B207	1.03E-09	1.5 %	RPV→ Refuelling floor→ Env. SFP→ Refuelling floor→ Env.	248	Late	VI	VI	BO-SDO-L BO-SFP_LOCA 02
LDHR-S09	7.85E-10	1.1 %	RPV→ D/W (cons. LOCA) → RB (through hatches etc.)→ Env.	3.94	Late	I	Ia	BO-SDO-L

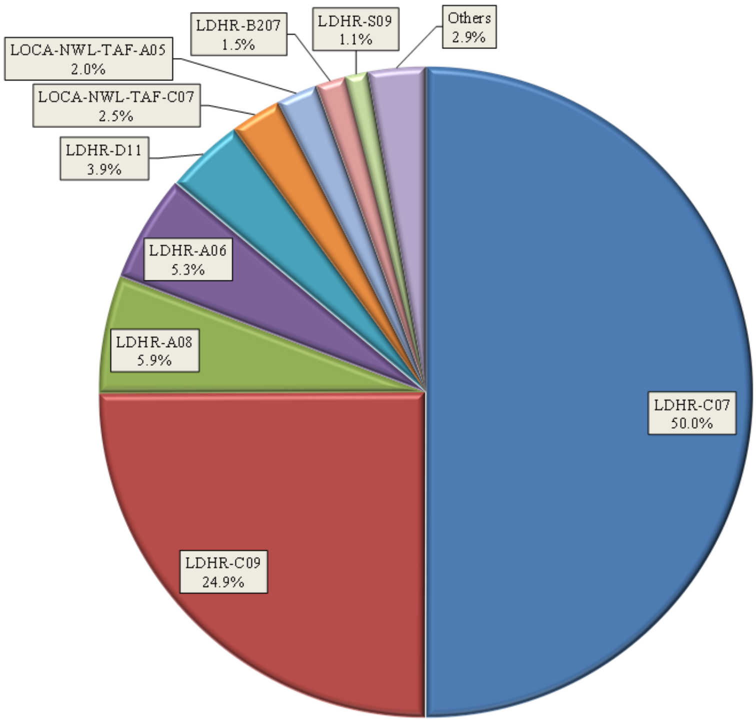

Figure 25.8.8-2 Contribution to LRF by Accident Sequences

(4) Significant PDSs

Table 25.8.8-3 shows the detailed contribution of PDSs to LRF.

The frequencies of PDSs III, IV, V, IV and IIV are equal to the frequencies of Accident Classes III, IV, V, VI and VII, respectively due to their definitions. The sum of the PDS frequencies Ia and Ib is equal to the sum of the Accident Classes I and II frequencies.

The highest contribution is PDS Ia: Fuel damage due to boil-off at low pressure (with S/P bypass) with late fission product release. This PDS represents 94.6 percent of the LRF. This is because:

Most of the fuel damage sequences are low pressure. A very limited number of POS S and POS D sequences, i.e., success of RCPB protection followed by failure of RPV depressurisation, end with high pressure. This results in small contribution of high pressure sequence (II, V).

Most of the fuel damage sequences are boil-off sequences. Even for a LOCA below TAF, the end states are assigned to PDS I if onset of boiling is expected before fuel uncovery. This results in small contribution of draindown sequence (III).

S/P availability for fission product scrubbing is examined (credited) only for non-LOCA events during POSs S, D. This results in small contribution of S/P available sequences (IV, V).

Dominant sequences in terms of frequency in Accident Class I are initiated by the loss of decay heat removal IEs (including LOOP) which have relatively higher frequencies than LOCA frequencies, such that PDS Ia (late fission product release) has much higher frequency than that of PDS Ib (early fission product release). This results in small contribution of early boil-off sequences (Ib).

Table 25.8.8-3 LRF Contribution by PDS

Rank	PDS	Frequency	Contribution
1	Ia	6.52E-08	94.6%
2	Ib	1.72E-09	2.5%
3	VI	1.38E-09	2.0%
4	VII	3.26E-10	0.5%
5	IV	2.53E-10	0.4%
6	V	6.89E-13	0.0%
7	II	0.00E+00	0.0%

25.8.8.3 Importance Analysis

Since the Level 2 analysis is based on the cutsets derived by the Level 1 analysis (with removal of the cutsets for Class VIII), the investigations performed in the Level 1 PSA [Ref-25.75] (see Section 25.8.6.3) are also applicable to the Level 2 PSA.

25.8.8.4 PSA Insights

Since the Level 2 analysis is based on the cutsets derived by the Level 1 analysis (with removal of the cutsets for Class VIII), the insights obtained from the Level 1 PSA (Section 25.8.6.4) are also applicable to the Level 2 PSA.

Additional discussions from the Level 2 PSA point of view are as follows.

- POS C has the highest contributor to the total LRF. All of the sequences for POS C are assigned to PDS Ia, Ib or III and the release categories with relatively high release amount. Reduction of FDF or LRF from POS C is important to reduce the offsite risk.

- POS B-2 has the second highest contributor to the total LRF. All the sequences for POS B-2 are linked to the release categories with relatively high release amount. Reduction of FDF or LRF from POS B-2 is important to reduce the offsite risk.

- Contributions from draindown type initiating events, e.g., LOCA below TAF, spurious RPV draindown, to the LRF are small. For most of the LRF scenarios, the RPV water level gradually decreases by boil-off when and after the fuel uncovery occurs. The time from the onset of fuel uncovery to the significant fission product release depends on the POS, i.e., decay heat. Although RPV injection after the onset of fuel uncovery is not credited in the Shutdown Level 2 PSA due to the characteristics of MAAP code [Ref-25.77], the RPV injection is possibly effective for accident mitigation in the actual accident progression. There is potential risk benefit and risk increase by this delayed injection. If sufficient amount of water is injected before the onset of actual fuel damage, fission product release would be prevented or significantly reduced. This is owing to the fission product scrubbing inside the RPV and prevention of RPV breach. This is especially beneficial when the scrubbing by the S/P is not credible, e.g., open RPV head condition, LOCA condition. Since the timing of delayed injection is highly uncertain, the risk increase by the delayed injection (after the significant fission product release) should be also studied. Possible scenarios are acceleration of metal-water reaction and acceleration of fission product entrainment by the generated steam.

25.8.9 Analysis for Level 3 PSA

In the SAA for IE at Shutdown, it is assumed that the containment boundary (blowout panel) is open throughout the transient, and that the Drywell Sprays and Lower Drywell Flooder System (LDF) fail to provide fission product transport mitigation function. Thus, unlike IE at Power, CETs are not developed for IE at Shutdown; rather each RC is directly linked to one of the PDSs identified at the interface between the Level 1 and Level 2 analyses. It is noted that this simplifying assumption leads to conservatively high releases to the environment for all Shutdown RCs.

The POSs considered in the Level 2 PSA [Ref-25.77] are:

- POS S: Transition to reactor cold shutdown,
- POS A: Transition to reactor disassembly and opening of the reactor well / pool gate,
- POS B-1: Full water level in reactor well and gate open with Division 2 in maintenance,
- POS B-2: Full water level in reactor well and gate open with Division 1 and 3 in maintenance,
- POS C: Transition to closed condition of the PCV head and RPV head, and
- POS D: Preparation of plant start up.

25.8.9.1 Radionuclide release categories

Six RCs have been defined in the IE at Shutdown Level 2 PSA. The RCs, the representative SAA sequences and their frequencies are given in Table 25.8.9-1. The allocation of RCs to Level 3 PSA cases, designated S1 to S6 is also shown. It is noted that in the case of IE at Shutdown, different designations are used for the Level 2 RCs and Level 3 PSA cases (unlike IE at Power where the RCs have the same designations as the Level 3 PSA cases).

For case S6, a combined condition is considered where releases from a spent fuel pool accident condition (release category ‘BO-SFP_LOCA 02’ in the SFP IE case ‘F1’) are added to the IE at Shutdown case S1 (release category BO-SDO-L).

Table 25.8.9-1 RCs, Level 3 PSA cases, PDSs, Representative SAA sequences and frequency of occurrence for IE at Shutdown

Level 3 PSA case	RC	POS	Representative SAA sequence	RC frequency
S1	BO-SDO-L	A, B, C	LOCA at RHR suction line outside PCV: RHR piping line break	6.69E-08
S2	BO-SDO-E	A, B, C	LOCA (mechanical) below Top of Active Fuel (TAF): bottom drain piping break	1.72E-09
S3	DD-SDO	S, D	RPV rupture during hydrostatic test	2.00E-11
S4	BO-SDL	S, D	Loss of decay heat removal (with depressurisation)	2.53E-10
S5	BO-SDH	S, D	Loss of decay heat removal (failure to depressurise)	1.03E-12
S6	BO-SDO-L + BO-SFP_LOCA 02	A, B, C	SFP Boil-off as a result of loss of RHR	1.38E-09

(1) Mass fraction releases and release profiles

Mass release predictions were generated for 12 fission product groups for the IE at Shutdown Level 3 PSA cases S1 to S5, consistent with those used for IE at Power (Section 25.6.1.1).

In the IE at Shutdown calculations, mitigation provided by the SGTS filtration system is not credited due to the likely degradation in conditions of severe accident. As mentioned previously, the containment boundary and the blow out panel are assumed to be open throughout the transient, providing a direct release path to the environment.

(2) Releases of radioactivity to the environment (source term)

In the IE at Shutdown (Base Cases), release amount is used directly with the reference core inventory, taking account only for the minimum radioactive decay period for each release category. Radionuclide releases to the environment were determined from the SAA mass release predictions using the same approach as for IE at Power (Section 25.6.1.1). It is noted that for RCs BO-SDH and DD-SDO (cases 5 and 3), some of the fuel is considered to have been transferred to the SFP; therefore the core inventory available for release is conservatively assumed to be 75 percent of the total.

In addition to the IE at Shutdown (Base Cases), which is considered to represent conservative decay conditions, two additional sensitivity studies were performed:

- The first sensitivity study, IE at Shutdown (Expanded Cases), takes account of the radioactive decay times associated with each POS. Radioactivity decay/ingrowth factors are applied to the inventory for the different POSs. For cases where the POS definition implies fuel has been off loaded into the SFP, the inventory available for release is scaled to 75 percent of the full core value.

- The second sensitivity study relates to release category BO-SDO-L (Level 3 PSA case S1) and considers the effect of the reduced decay heat inherent in POS C in the SAA. It considers the combined impact of reduced decay heat on the accident progression, the predicted FREL[n] values and the additional radioactive decay inherent in POS C.

The releases are split into phases in a similar manner to IE at Power (Section 25.6.1.1); the only difference being that the MAAP 4.07 transient is run for 96 hours for the Base Cases and up to 720 hours for the Reduced Decay Heat (RDH) sensitivity cases.

25.8.10 Results (Shutdown Level 3)

25.8.10.1 Facility Dose Bands

The doses for each Level 3 PSA case for IE at Shutdown are calculated at distances of 400 m, 700 m, 1,000 m and 1,500 m. The facility dose band allocations are determined based on the highest summated long term dose (cloud gamma, inhalation and long term ground gamma) for each case.

Table 25.8.10-1 presents the assessment against facility dose bands for the IE at Shutdown. It can be seen that all the release categories lie in the > 1 Sv dose band. This reflects the conditions in the SAA where no credit is claimed for filters and the containment boundary is assumed to be not intact (and the blow out panel is assumed to be open) throughout the transient. The summated frequency contribution is 7.03E-08 /y or 7 percent of the BSO.

Three release categories contribute almost all of the frequency in the highest facility dose band:

- Release category BO-SDO-L is the largest contributor at 6.69E-08 /y or 95.2 percent of the total,
- Release category BO-SDO-E contributes 1.72E-09 /y or 2.4 percent of the total, and
- The combination case {BO-SDO-E + BO-SFP_LOCA 02} contributes 1.38E-09 /y or 2.0 percent of the total.

Table 25.8.10-1 Assessment against Facility Dose Bands for the IE at Shutdown

Facility Dose Band (Sv)	Release categories assigned to each dose band for IE at Shutdown leading to fuel melt	Band Frequency (/y)	Percentage of Band BSO	BSO (/y)	BSL (/y)
0.0001 to 0.001	-	-	-	1.00E-02	1.00E-01
0.001 to 0.01	-	-	-	1.00E-03	1.00E-01
0.01 to 0.1	-	-	-	1.00E-04	1.00E-02
0.1 to 1	-	-	-	1.00E-05	1.00E-03
> 1	BO-SDO-L, BO-SDO-E, DD-SDO, BO-SDL, BO-SDH, {BO-SDO-E + BO-SFP_LOCA 02}	7.03E-08	7.0 %	1.00E-06	1.00E-04
Summated frequency of food bans /y		7.03E-08			

25.8.10.2 Individual Risk

Conditional individual risks are calculated for each Level 3 PSA case for IE at Shutdown at distances of 400 m, 1,000 m and 1,500 m from the site to illustrate how the conditional individual risk varies with distance over the range of likely offsite locations with continuous occupancy. Again, these results assume minimal offsite protective actions.

The conditional individual risks given in the PSA Summary Report [Ref-25.1] are the sum of the conditional risk of early (deterministic) fatal health effects and conditional risk of notional late (stochastic) health effects leading to a fatality. For all the IE at Shutdown, the individual risk of early fatal health effects is at least half of the total individual risk. The aggregation of conservatisms in the source term modelling in the SAA, the bounding approach used in the Level 2 / Level 3 interface and the assumption of minimal offsite protective actions is thought to lead to this significant risk contribution from short term fatal health effects.

For the combination event in Level 3 PSA case S6 {BO-SDO-E + BO-SFP_LOCA 02}, the conditional individual risk increases only a small amount compared to case S2 {BO-SDO-E} alone. This is thought to be an artefact of the use of a single release phase.

The individual risk for each Level 3 PSA case for IE at Shutdown is calculated as the product of the conditional individual risk at the three distances given in the PSA Summary Report [Ref-25.1] and the release category frequency given in Table 25.8.9-1. The contribution of each release category to the overall risk from IE at Shutdown is given in Table 25.8.10-2 and Figure 25.8.10-1.

Table 25.8.10-2 Individual Risk of Fatality near to the Site for the IE at Shutdown with Minimal offsite Protective Actions

Release Category	Release Category Frequency (/y)	Individual risk of fatal health effects (/y)			Contribution to Total Risk at 1 km
		400 m	1,000 m	1,500 m	
BO-SDO-L	6.69E-08	1.65E-08	1.23E-08	1.06E-08	96.0 %
BO-SDO-E	1.72E-09	4.49E-10	2.97E-10	2.25E-10	2.32 %
DD-SDO	2.00E-11	5.02E-12	3.13E-12	2.30E-12	0.02 %
BO-SDL	2.53E-10	4.17E-11	2.40E-11	1.72E-11	0.19 %
BO-SDH	1.03E-12	1.01E-13	3.56E-14	1.95E-14	<0.01 %
{BO-SDO-E + BO-SFP_LOCA 02}	1.38E-09	2.46E-10	1.87E-10	1.68E-10	1.46 %
Total individual risk: /y		**1.72E-08**	**1.28E-08**	**1.10E-08**	
Total as % of BSO		**1.72 %**	**1.28 %**	**1.10 %**	
BSO		**1.00E-06**			
BSL		**1.00E-04**			

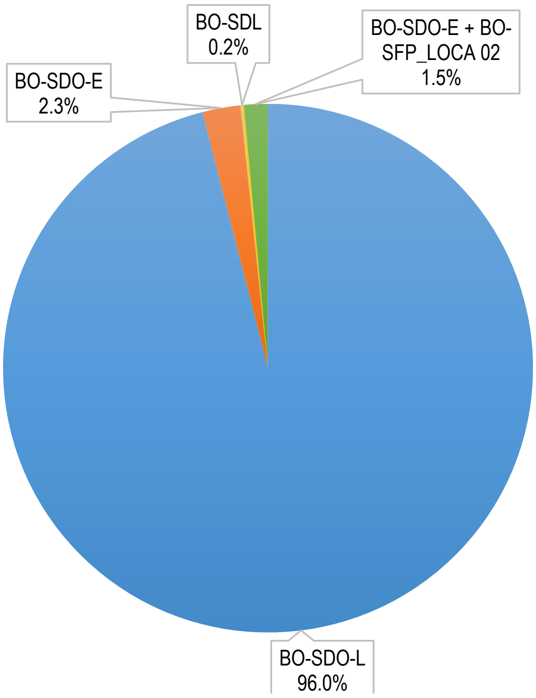

Figure 25.8.10-1 Contribution of release categories to the individual risk at 1 km for IE at Shutdown leading to fuel melt

The summated individual risk at 1 km from IE at Shutdown is 1.28E-08 /y or about 1.3 percent of the BSO. It can be seen that three release categories contribute almost all of the individual risk:

- Release category BO-SDO-L is the largest contributor at 1.23E-08 /y or 96.0 percent of the total, and is individually about 1 percent of the BSO.
- Release category BO-SDO-E contributes 2.97E-10 /y or 2.3 percent of the total.
- The combination case {BO-SDO-E + BO-SFP_LOCA 02} contributes 1.87E-10 /y or 1.5 percent of the total. This release is currently treated as a single phase and its contribution to individual risk would be likely to increase if it was able to be treated as a multiple phase release. However, the frequency contribution is low at 1.38E-09 /y, so more realistic treatment of release profile would be unlikely to significantly increase the total individual risk.

25.8.10.3 Societal Risk

The summated mean numbers of health effects predicted in the UK population for IE at Shutdown cases are calculated. This is accompanied by a breakdown between the mean number of early fatal health effects and the mean number of notional late fatalities. The conditional probability of exceeding 100 notional late fatalities is also given and it can be seen that these are close to unity in all the Level 3 PSA cases.

All the conditional results presented are conservative values based on minimal offsite protective actions; however, this is not thought to affect the assessment against the societal risk criterion of Target 9.

Table 25.8.10-3 and Figure 25.8.10-2 present the assessment against Target 9 for the IE at Shutdown. This is based on the summated numbers of short term fatal health effects and notional late fatalities in the UK population presented in the PSA Summary Report [Ref-25.1]. The summated societal risk from IE at Shutdown is 7.03E-08 /y, i.e., about 70 percent of the BSO.

It can be seen that three release categories contribute almost all of the frequency above the societal risk threshold:

- Release category BO-SDO-L is the largest contributor at 6.69E-08 /y or 95.2 percent of the total.
- Release category BO-SDO-E contributes 1.72E-09 /y or 2.4 percent of the total.
- The combination case {BO-SDO-E + BO-SFP_LOCA 02} contributes 1.38E-09 /y or 2.0 percent of the total.

Table 25.8.10-3 Societal consequences in the UK population for IE at Shutdown with minimal offsite protective actions

Release Category	Release category frequency (/y)	Frequency above Target 9 threshold (/y)	Contribution to Total Frequency
BO-SDO-L	6.69E-08	6.69E-08	95.2 %
BO-SDO-E	1.72E-09	1.72E-09	2.45 %
DD-SDO	2.00E-11	2.00E-11	0.03 %
BO-SDL	2.53E-10	2.53E-10	0.36 %
BO-SDH	1.03E-12	1.03E-12	<0.01 %
{BO-SDO-E + BO-SFP_LOCA 02}	1.38E-09	1.38E-09	1.96 %
	Total Frequency: /y	**7.03E-08**	
	Total as % of BSO	**70.3 %**	
	BSO	1.00E-07	
	BSL	1.00E-05	

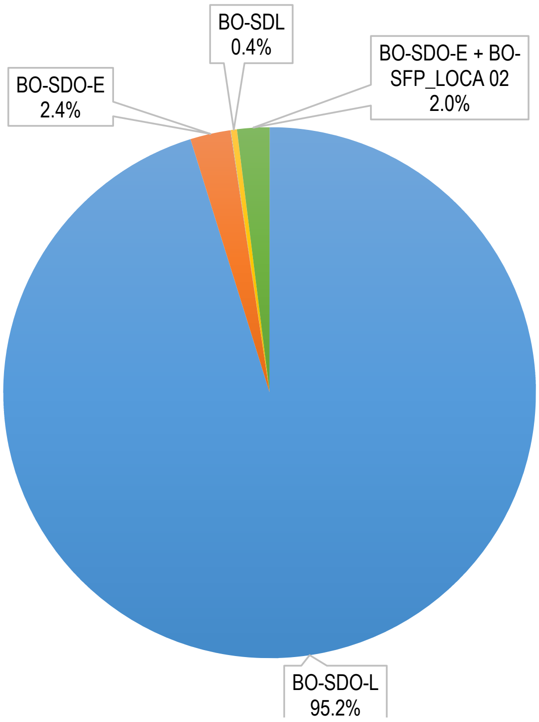

Figure 25.8.10-2 Contribution of release categories to the frequency of exceeding the societal risk criterion for IE at Shutdown leading to fuel melt

25.8.11 Uncertainty and Sensitivity Analysis

25.8.11.1 Sensitivity Analysis

Sensitivity analyses have been performed for each key safety function using quantitative and/or qualitative methods (see Section 25.7.5.2 (b)).

Table 25.8.11-1 to 25.8.11-3 shows the summary of the sensitivity analyses.

Table 25.8.11-1 Sensitivity Analysis for Fuel Cooling

No.	Sensitivity Analysis Case	Difference		Remarks
		FDF	LRF	
1	Effect of rector well overfill	2.5 %	3.2 %	Small impact on the PSA results.
2	Adverse effect on equipment inside reactor building following BOC	16 %	20 %	Significant impact on LRF. All divisions of safety systems are assumed unavailable due to the effect of flooding in this sensitivity analysis. This would be a conservative analysis because the success of BOC isolation would prevent the all division failures.
3	Credit of fire protection system to SFP injection	-21 %	-1.8 %	Significant impact on the PSA results.
4	Adverse effect on equipment inside reactor building following LOCA above NWL	720 %	930 %	Significant impact on the PSA results. LOCA above NWL includes the heavy load drop which is the main contributor. The impact of heavy load drop is assessed in [Ref-25.78] which describes the claim as following. *"A Design Basis RPV Top Head drop will not compromise delivery of the Fundamental Safety Functions (FSFs). This is taken to mean that no unacceptable Design Basis fault is caused by the RPV Top Head drop."* Therefore this assumption of the sensitivity analysis would be conservative.
5	Definition of global fuel damage	-1.5 %	-1.9 %	Small impact on the PSA results.
6	Time margin during LOCA	0.0 %	0.0 %	Minimum impact on the PSA results.
7	Multiple demand for RPV water level control	0.7 %	0.8 %	Minimum impact on the PSA results.

Table 25.8.11-2 Sensitivity Analysis for RCPB protection

No.	Sensitivity Analysis Case	Difference		Remarks
		FDF	LRF	
8	Re-evaluate CCF probability of SRVs for passive RPV overpressure	-28 %	-36 %	Significant impact on the PSA results.
9	Refinement of POS A and POS C	-17 %	-22 %	Significant impact on LRF.
10	Effect of consequential LOCA	-26 %	-32 %	Significant impact on the PSA results.
	Combined case of No.8, 9 and 10	-29%	-36 %	Significant impact on the PSA results.
11	Credit of SRV, RPV top vent and MSIV opening	-28 %	-35 %	Significant impact on the PSA results.

Table 25.8.11-3 Sensitivity Analysis for Others

No.	Sensitivity Analysis Case	Difference		Remarks
		FDF	LRF	
12	LOOP frequency and non-restoration probability during shutdown	10 %	13 %	Small impact on the PSA results.
13	LOCA frequency	-3.6 %	-4.3 %	Small impact on the PSA results.
14	Recovery of failed component	-49 %	-40 %	Significant impact on the PSA results.
15	LOCA below TAF caused by dropped load	0.0 %	0.0 %	Minimum impact on the PSA results.
16	Water shortage in case of min flow from CST to S/P	0.1 %	0.1 %	Minimum impact on the PSA results.
17	Dependency of Multiple Human Failure Events (HFEs) (Lower Dependency)	-0.1%	-0.1 %	Minimum impact on the PSA results.
	Dependency of Multiple Human Failure Events (HFEs) (Higher Dependency)	1,200 %	63 %	Significant impact on the PSA results. Common cognition errors are introduced as described in HRA report Rev. E [Ref-25.44]. The sensitivity analysis is performed in No.21.
18	RPV water level control at LOCA/BOC	3.8%	4.7 %	Small impact on the PSA results.
19	Addition of CCF between EDG and BBG	5.3 %	6.6 %	Small impact on the PSA results.
20	Detection rate by self-testing for C&I digital systems (undetectable rate is 5 %)	-7.8 %	-4.7 %	Small impact on the PSA results and increased margin compared with NSEDP/SAP risk targets.
	Detection rate by self-testing for C&I digital systems (undetectable rate is 20 %)	17 %	10 %	Small impact on the PSA results.
21	Updated HRA report (revision E)	201 %	19 %	Significant impact on the PSA results.

25.8.11.2 Uncertainty Analysis

Uncertainty analyses were performed for supporting the robustness of the PSA and design of the UK ABWR. The purpose of the uncertainty analyses is to show that the mean values of FDF or LRF resulting from the uncertainty propagation is below the NSEDP/SAP targets.

Uncertainty analysis was performed using the Monte Carlo sampling method which generates a probability density function and a cumulative probability function for Fuel Damage Frequency (FDF), Large Release Frequency (LRF) and Large Early Release Frequency (LERF).

A sample size of 100,000 was used to generate the associated results from the cutsets with the truncation value of 1.0E-14 /y from which FDF and LRF converge. The mean values generated are FDF of 1.05E-07 /y, LRF of 8.86E-08 /y and LERF of 2.00E-9 /y as shown in Tables 25.8.11-4 to 25.8.11-6. The results for each POS and Release Category are shown in Tables 25.8.11-7 to 25.8.11-10.

Mean values of FDF, LRF and LERF are higher than point-estimated values by about 10 to 30 percent.

Table 25.8.11-4 Uncertainty Analysis for Total FDF

Case	Mean	5 %	Median	95 %
FDF	1.05E-07	7.29E-09	4.59E-08	3.15E-07

Table 25.8.11-5 Uncertainty Analysis for Total LRF

Case	Mean	5 %	Median	95 %
LRF	8.86E-08	4.41E-09	3.42E-08	2.68E-07

Table 25.8.11-6 Uncertainty Analysis for Total LERF

Case	Mean	5 %	Median	95 %
LERF	2.00E-09	5.63E-11	5.49E-10	7.62E-09

Table 25.8.11-7 Uncertainty Analysis for FDF by POS

Case	Mean	5 %	Median	95 %
FDF (POS S)	1.80E-09	3.62E-11	3.97E-10	4.88E-09
FDF (POS A)	1.58E-08	7.94E-10	4.81E-09	3.98E-08
FDF (POS B-1)	6.03E-09	6.09E-11	7.70E-10	1.89E-08
FDF (POS B-2)	1.51E-08	3.06E-10	2.91E-09	5.07E-08
FDF (POS C)	6.96E-08	1.97E-09	2.21E-08	2.07E-07
FDF (POS D)	5.41E-09	5.89E-11	8.70E-10	1.32E-08

Table 25.8.11-8 Uncertainty Analysis for LRF by POS

Case	Mean	5 %	Median	95 %
LRF (POS S)	1.81E-09	3.62E-11	4.00E-10	4.89E-09
LRF (POS A)	1.45E-08	7.93E-10	4.77E-09	3.94E-08
LRF (POS B-1)	8.12E-10	8.57E-12	1.16E-10	2.37E-09
LRF (POS B-2)	1.89E-09	2.23E-11	2.97E-10	5.54E-09
LRF (POS C)	6.73E-08	1.97E-09	2.22E-08	2.10E-07
LRF (POS D)	4.41E-09	5.74E-11	8.63E-10	1.33E-08

Table 25.8.11-9 Uncertainty Analysis for LERF by POS

Case	Mean	5 %	Median	95 %
LERF (POS S)	3.67E-11	3.46E-14	4.61E-12	1.56E-10
LERF (POS A)	1.54E-09	1.80E-12	2.12E-10	6.56E-09
LERF (POS C)	4.07E-10	1.89E-11	1.40E-10	1.31E-09
LERF (POS D)	6.27E-12	4.18E-15	2.98E-13	1.85E-11

Table 25.8.11-10 Uncertainty Analysis for Release Category Frequency

Release Category	Mean	5 %	Median	95 %
BO-SDH	1.71E-12	3.05E-15	1.39E-13	5.66E-12
BO-SDL	2.71E-10	1.28E-11	1.16E-10	9.45E-10
DD-SDO	1.97E-11	1.39E-14	8.57E-13	5.33E-11
BO-SDO-E	2.04E-09	5.08E-11	5.33E-10	7.69E-09
BO-SDO-L	9.31E-08	3.90E-09	3.24E-08	2.62E-07
BO-SFP_LOCA_02	2.42E-09	4.21E-11	3.97E-10	6.49E-09

25.8.12 Insights of Sensitivity Analysis and Uncertainty Analysis

The most significant insights from the results of the sensitivity and uncertainty analyses are discussed here. Three sensitivity analyses which resulted in an increase of Shutdown FDF and/or LRF by 20 percent or more are summarised in Table 25.8.12-1.

Overall, it was found that risk impacts by most of the sensitivity cases are not significantly increased except for three cases as described below.

Sensitivity cases which are increased from the base case by roughly 20 percent or more are identified as shown in the table below. These cases are considered in further assessments as described below.

- Sensitivity case No.1 identified that the assumption of an affected division due to BOC had a large impact on the Shutdown LRF. At first, the impact of BOC is confirmed to check the results of the flooding PSA during shutdown. If the impact is identified as significant, the operator action for BOC isolation will be examined in order to develop the Tabular Task Analysis for HEP evaluation.

- Sensitivity case No.2 identified that the adverse effect of steam during LOCA had a large impact on the Shutdown FDF and LRF. The heavy load drop is the major contributor. The frequency of heavy load drop will be examined in order to develop the TTA for HEP evaluation.

- Sensitivity case No.3 identified that the HEP dependency is an important consideration for the Shutdown PSA. The adequacy of the HEP dependency was reviewed considering the result of this sensitivity analysis.

- The mean values generated by the uncertainty analysis are Shutdown FDF of 1.05E-07 /y and Shutdown LRF of 8.86E-08 /y.

Table 25.8.12-1 Sensitivity Analysis Results Increased from the Base Case by 20 percent or More

No.	Sensitivity Analysis Case	Difference		Remarks
		FDF	LRF	
1	Adverse effect on equipment inside reactor building following BOC	16 %	20 %	Significant impact on Shutdown LRF. All divisions of safety systems are assumed unavailable due to the effect of flooding in this sensitivity analysis. This is a conservative analysis because the success of BOC isolation would prevent the all division failures.
2	Adverse effect on equipment inside reactor building following LOCA above NWL	720 %	930 %	Significant impact on the Shutdown PSA results. LOCA above NWL includes the heavy load drop which is the main contributor. The impact of heavy load drop is assessed in [Ref-25.78] which describes the claim as following. “A Design Basis RPV Top Head drop will not compromise delivery of the Fundamental Safety Functions (FSFs). This is taken to mean that no unacceptable Design Basis fault is caused by the RPV Top Head drop.” Therefore this assumption of the sensitivity analysis would be conservative. Furthermore, the heavy load drop frequency is simply estimated in [Ref-25.78]. The frequency will be re-estimated if the detailed analysis is needed.
3	Dependency of Multiple Human Failure Events (HFEs) (Higher Dependency)	1,200 %	63 %	Significant impact on the Shutdown PSA results. This would be a bounding analysis. During POS B-1 and B-2, time margin is over 10 hours in some initiating events.

25.8.13 Insights from Assessment

Insights from Shutdown PSA are described below.

- POS C has the highest risk contribution because Class 1 Divisions 1 and 3 systems are out of service. This is a natural insight clearly related to the assumed outage schedule defined in Section 25.8.2. Loss of AC power results in only FLSS and FLSR being available for makeup. Modification of the outage schedule was recommended for future improvement by the experienced system engineer and outage planner in the Design Review meeting. A sensitivity analysis was performed especially focusing on the dominant sequence: RPV pressurisation and failure to open all SRVs (safety valve function).

- POS B-2 is the second highest contributor because only Class 1 Division 2 systems are available like POS C, automatic initiation of makeup system is not available, and unique IEs challenging the SFP exist, e.g., LOCAs above NWL (dominated by heavy load drop) and LOCAs at the SFP (dominated by heavy load drop). It was recommended at the Design Review meeting on November 2015 that significant time margins in POSs B-1 and B-2 should be adequately taken into account for future PSA improvement. A sensitivity analysis was performed focusing on the component recoveries of selected systems.

- Due to the partially degraded redundancy of the Class 1 systems, FLSS is important during outages as well as at Power. The capability of FLSS to provide SFP makeup also makes FLSS important during outages. The post-initiator HFE that has the highest F-V and RAW is the manual initiation of FLSS.

- FLSR also provides considerable risk reduction in outages. It is assumed that FLSR is required to be connected to the injection point before an IE during POS B-2 and POS C per the recommendation by the experienced system engineer and maintenance planner in the Design Review meeting, such that redundancy of mitigation systems is improved under the condition in which Class 1 Divisions 1 and 3 systems are in maintenance.

- Risk contribution by CCF events is generally lower than those in the at Power PSA. That is because system or component redundancy is degraded during an outage compared to at Power conditions.

- Pre-initiator HFEs are not very risk significant, which is consistent with the at Power PSA.

- Initiating event HFEs (Type B) have relatively small risk contribution (up to about 2 percent of total FDF), except for LOCA (mechanical) above normal water level which is dominated by heavy load drop frequency and has a high frequency (about 2E-03 /y).

- Availability of RPV injection paths given overpressure failure of the reactor coolant pressure boundary has large impact on the shutdown risk. A sensitivity analysis was performed especially focusing on the dominant sequence: RPV pressurisation and failure to open all SRVs (safety valve function).

- POS C has the highest contributor to the total LRF. All the sequences for POS C are assigned to PDS Ia, Ib, II or III and the release categories with relatively high release amount. Reduction of FDF or LRF from POS C is important to reduce the offsite risk.

- POS B-1 has the second highest contributor to the total LRF. All the sequences for POS B-1 are linked to the release categories with relatively high release amount. Reduction of FDF or LRF from POS B-1 is important to reduce the offsite risk.

25.8.14 Key Assumptions and Study Limitations

Assumptions in the Shutdown PSA were made in the development phase. They relate to each aspects of PSA, such as the identification of initiating events, the definition of success criteria, reliability data selection, containment performance, accident progression and so on. In addition, when the level of plant requirements specification limits the detail of the PSA, then appropriate assumptions were developed to allow completion of the PSA.

Starting from assumptions, sensitivity analyses have been performed. Key assumptions, which have a comparatively large impact on the results, have been listed from the result of sensitivity analyses.

The key assumptions considered in the Shutdown PSA are listed below.

- BOC is assumed to cause loss of systems in the same division of the R/B. Survivability of R/B systems given a BOC is uncertain.

- The fire protection system is not credited in this PSA. The additional credit of the fire protection system for SFP cooling reduces the risk effectively as shown in Table 25.8.11-1 (Case No.3).

- The steam may be released from the containment to the Reactor Building (R/B) during LOCA inside containment. This effect on equipment in the R/B is uncertain.

- The following assumptions in terms of the consequential LOCA are made.

  - Failure to open all SRVs (safety valve function) after onset of evaporation results in consequential LOCA (small leak from SRV valve sheets) during POSs S, A, C and D for loss of decay heat removal events or LOCAs after isolated. This assumption is based on the expectation that the most likely leak location is the SRV valve sheets given overpressurisation.

  - Given a consequential LO CA, the leak rate from the SRV valve sheet(s) is balanced with the evaporation rate, such that the RPV pressure is kept higher than the HPCF pump head pressure and thus HPCF is not credited.

  - Given a consequential LOCA, subsequent RPV depressurisation by SRV relief valve function, ADS or RDCF are not credited since the safety valve function has already failed. As such, the consequential LOCA is treated as fuel damage in the event tree.

- Recovery actions such as restoration of RHR etc. are not credited because it is difficult to identify a failure mode of the failed component.

- Dependency levels among multiple Human Failure Events are based on NUREG-1921 [Ref-25.79].

25.9 Spent Fuel Pool PSA

25.9.1 Scope of SFP PSA

Fuel cooling conditions are similar between SFP and shutdown conditions of reactor core (with full water level in the reactor well) in terms of;

- Large water inventory at the start of accident,
- Small decay heat compared to the reactor core at Power, and
- Similarity in cooling systems.

Considering the above similarity, Hitachi-GE uses a similar approach/methodology for the internal events SFP PSA to that for internal events shutdown PSA for the reactor (Section 25.8).

<u>SFP PSA Plant Operational States</u>

The SFP PSA encompasses six plant operating states (POSs) namely:

- POS F: Reactor at full power or low power states,
- POS S: Transition to reactor cold shutdown,
- POS A: Transition to reactor disassembly with Division 2 in maintenance,
- POS C: Transition to closed condition of PCV/RPV heads with Divisions 1 and 3 in maintenance,
- POS D: Preparation of plant startup, and
- POS E: Full core off-loaded to the SFP.

In the above POSs the SFP is isolated from the reactor well. SFP risk is included in the Shutdown PSA (Section 25.8) when the reactor well gate is open.

25.9.2 Initiating Events

The main objectives of the IE analysis for internal events SFP PSA are:

- Investigate and identify a reasonably complete set of the events that potentially lead to fuel damage,
- Prioritise/group them, and
- Provide estimates for the frequencies of the initiating event groups using information available and associated estimation techniques.

25.9.2.1 Reference and Scope of IE analysis for SFP PSA

The IE analysis is conducted based on the design and operational features of UK ABWR described in the Generic PCSR and so on. Designs of SFP and SFP-related systems are summarised in [Chapter 19 of PCSR]. SFP PSA covers the at Power condition and shutdown mode.

25.9.2.2 IE Analysis

Section 25.6.7 of the AESJ Standard [Ref-25.80] recommends the following approaches/methodologies for IE frequency estimation.

- Generic data and/or operational experience (if judged applicable), only used for LOOP,
- IE frequency used for at Power PSA (if judged applicable),
- Logic model (fault tree analysis, event tree analysis and/or human reliability analysis), e.g. support system initiators,
- Engineering judgement (e.g. expert elicitation process on RPV rupture frequency), and
- Converting IE frequencies to per year basis.

IE frequency estimation for UK ABWR SFP PSA is based on the failure rates of related SSCs and/or human reliability analysis, except for LOOP, Loss of SFP Liner and SFP Gate / Seal failure.

Each IE of the UK ABWR PSA is quantified on a per calendar year basis. In the SFP PSA, the following common approach was taken to derive the initiating event frequency.

- Initiating event frequencies for the at Power condition (POS F) were calculated on a calendar year basis. An availability factor of 0.9 was used to convert any reactor critical year frequencies to calendar year frequencies.
- To avoid omission or double-counting of internal events risk, it is assumed that the SFP PSA covers 0.1 (=1.0-0.9) of a calendar year in average in shutdown condition. This is consistent with reactor side PSA on reactor fault. For the generic data provided as reactor critical year basis, e.g. LOOP, the reactor critical year frequency is multiplied by 0.1 as the “total” IE frequency for shutdown condition (POS S to POS D). That “total” IE frequency is further allocated to the relevant POSs as needed using the period of each POS as weighting factor. The weighting factors are as follows.
  - POS S: 1 d / 27.5 d = 0.0364
  - POS A: 2 d / 27.5 d = 0.0727
  - POS C: 4 d / 27.5 d = 0.146
  - POS D: 7.5 d / 27.5 d = 0.273
- To calculate the frequency of POS E (full core off load condition), Loss of RHR is considered as a Loss of SFP cooling initiating event. However, loss of FPC is not considered an initiating event. To quantify the IE frequency, POS E duration and available systems were defined. Following conditions are assumed in POS E:
  - Performance timing: Full core off load is not performed in regular refuelling outage. Long duration outages are performed to allow focussed inspection and occasionally, after forced shutdown, the full core may be removed from the reactor and stored in SFP. The period between full core off load outages are not planned but, based on the discussion with the plant field engineering team, a full core off load is assumed to occur once every five years.
  - Duration: As noted above, full core off load is not a planned extension of the outage duration. But 60 days has been assumed as the duration of the outage.
- It should be noted that the assumption (60 days every five years) has large uncertainty. In addition, the shutdown PSA also uses the weighting factors, except POS E which is specific to the SFP PSA.
- For the IE frequencies for shutdown condition derived from system fault trees or hand calculation based on hourly component failure rates, the frequency per hour is multiplied by 24[h] times 365[d], and then 0.1. That “total” IE frequency is further allocated to the relevant POSs using the period of each POS as weighting factor. That “total” IE frequency is further allocated to the relevant POSs using the duration of each POS as a weighting factor as previously described.

The SFP IE frequencies are summarised in Table 25.9.2-1. The individual IE frequencies are discussed in the subsections below.

(1) Loss of FPC

Faults which affect operation of FPC are grouped as initiating event of “Loss of FPC”. The total frequency for loss of FPC is estimated using fault trees. The necessary functions for FPC operation are:

- Continuous operation of FPC pump,
- Heat removal from FPC to RCW,
- Opening FPC line from and to SFP, and
- Avoiding diversion flow outside of FPC loop.

(2) Loss of Offsite Power

Loss of Offsite Power (LOOP) is treated separately from loss of decay heat removal IEs in order to capture the dependency of various systems on power. If Class 1 AC power supply by an EDG is established after LOOP, restart of the FPC, as well as its support systems, is required since they stop due to LOOP.

(3) Support System Initiator

Support system initiators are classified into the following groups:

- Support system initiators which are treated separately treated from “Loss of FPC”
- Support system initiators which are included in “Loss of FPC”
- Support system initiators which only affects FPC in standby condition (FPC-A)
- Support system initiators which lead to a loss of offsite power

Support system initiators which are treated separately treated from “Loss of FPC”

Loss of support systems that affect not only operation of FPC but also that of other front line systems in the SFP PSA are treated separately from “Loss of FPC”. The initiating event groups are classified as follows:

- Loss of Class 1 AC-D
- Loss of RCW-B/ RSW-B
- Loss of HVAC-R/BEEE/Z HVAC-B
- Loss of Class 3 AC-B1
- Loss of HVAC-Hx/B-N

For the “Loss of HVAC-Hx/B-N”, emergency Hx/B HVAC is initiated by high room temperature. Failures of the both HVAC trains lead to a loss of RCW. Therefore, “Loss of HVAC-Hx/B-N” is classified into “Loss of Support System Initiators”.

The approach and assumptions used for the PSA for reactor fuel at Power (Section 25.4.1) are applied for initiating event frequencies of these initiating event groups.

Support system initiators which are included in “Loss of FPC”

Some initiating event groups do not affect operation of the front-line systems in SFP PSA except the FPC itself. Therefore, they are included in “Loss of FPC”. The Initiating group classified into the “Loss of FPC” is the “Loss of IA/SA”.

Support system initiators which only affects FPC in standby condition (FPC-A)

Some initiating event groups do not affect operation of FPC-A but affect backup operation of FPC-A. Failure to initiate FPC-A is modelled in the “Loss of FPC” as a backup operation for FPC-B. Fault trees of support systems for FPC-B are linked to that of FPC-B in the same way as that for other front-line systems. Therefore, the following initiating events are considered in the fault tree of “Loss of FPC”:

- Loss of Class 1 AC-C
- Loss of RCW-A/ RSW-A
- Loss of HVAC-R/BEEE/Z HVAC-A
- Loss of Class 3 AC-A1

Support system initiators which leads to loss of offsite power
Some support system initiators lead to loss of Class 3 AC power similar to a loss of offsite power. If Class 1 AC power supply by an EDG is established after LOOP, restart of the FPC, as well as restart of its support systems, is required since they stop due to LOOP or Class 3 AC power. The following initiating events are considered:

- Loss of HVAC-TBNEE/Z
- Loss of HNCW

(4) Reactor Faults Impacting the SFP

Some reactor faults degrade the reactor building environment. These impact the SFP, SFP cooling or support systems in the R/B. Operator actions supporting SFP cooling may also be affected. Reactor events impacting the SFP are:

- LOCA, and
- Containment failure followed by
  - ATWS, or
  - Core damage event resulting in Containment leakage, Venting or Containment failure.

The above reactor challenges may result in the following:

- Release of fission products to the Reactor Building
- Release of hydrogen to the Reactor Building
- Release of steam to the Reactor Building

These effects would, in turn, affect the following:

- Access to the Reactor Building

- Recovery of equipment
- Environmental failure of equipment
    - Potential for SFP structural failure

These effects from Reactor Faults are considered as shown below:

- Containment failure sequence fault trees are linked to the starting point of the event tree as an initiating event.
- SFP structural integrity, including that of liner, is assumed to be lost due to hydrogen burning using an assumed conditional probability in certain containment failure modes. Piping failures due to hydrogen burning is also assumed. Different conditional probabilities are assumed for each containment failure mode.
- SFP structural integrity including that of liner is assumed to be lost due to strain of SFP structural material due to RCCV pressure with using assumed conditional probabilities in certain containment failure modes. Different conditional probabilities are assumed for each containment failure mode.
- SFP cooling and support systems in the R/B, such as FPC, RHR, RCW, emergency AC power supply, are assumed to be lost due to steaming in the R/B with assumed conditional probabilities in certain containment failure modes. In this case, make up from outside R/B, that is FLSS and FLSR, can be credited.
- Certain containment failure sequences with "no core damage" also release steam into R/B. SFP structural integrity, including that of liner, is assumed to be lost due to strain of SFP structure material by RCCV overpressure with an assumed conditional probability. SFP cooling or support systems in the R/B such as FPC, RHR, emergency AC power supply, are assumed to be lost but make up function from outside R/B, that is FLSS and FLSR, can be credited.

(5) Loss of SFP inventory

Identification of initiating events for SFP PSA lists possible initiators for "Loss of SFP inventory" as follows:

- Failure associated with RHR connections to SFP
- Equipment / crane falling into the SFP
- Dropped cask/heavy load drop
- Loss of Spent Fuel Pool Liner
- Siphoning of Spent Fuel Pool Inventory/ SFP draindown
- Spent fuel pool Gate / Seal failure

Failure associated with RHR connections to SFP

Failure associated with RHR connections to SFP may cause a loss of SFP inventory. Therefore, check valves and a siphon breaker are installed to reduce the probability of a loss of SFP inventory from a failure associated with RHR.

Equipment / Crane falling into the Spent Fuel Pool

There are two potential effects if a heavy object is dropped into the SFP. First, a sufficiently heavy object could damage SFP integrity, causing a drain down of the SFP. Second, the object could land on the fuel causing mechanical damage and release of fission products from the gap between the fuel and the fuel rod (noble gas and iodine) to reactor water. The fission products would be released to the environment through the reactor building. According to Design Basis Analysis, it is evaluated that amount of fission product is limited from the fuel failure and the influence to environment is very small [Ref-25.23]. This fuel damage frequency of this event is addressed as a part of Target 8 and addressed in Section 25.12.

Occurrence frequency of equipment / crane falling into the SFP and cask drop is estimated using an event tree analysis.

In the frequency estimation of initiating event analysis for SFP PSA, all the travel paths of lifting were considered.

Hydrogen combustion was considered in the conditional probability for the containment failure sequence frequency and the crane drop effect is bounded by this analysis.

The above approach and assumptions are the same as those applied to drop event frequency calculations for the shutdown PSA.

Dropped Cask

Occurrence frequency of cask drop into the SFP is estimated including Equipment / Crane falling.

Loss of Spent Fuel Pool Liner

The SFP Liner provides the leakage barrier for the SFP. Damage to the liner would allow leakage into the gap between the liner and the structure of the concrete wall and could result in draining of the SFP water. Loss of SFP inventory is considered to occur mainly by induced external force such as seismic induced failure. However, failure of the liner by static load is modelled in order to capture the importance of this event in the SFP PSA.

The frequency of loss of inventory events in which loss of more than 1 foot (about 0.3 m) of water occurred is estimated in NUREG-1275 vol.12 [Ref-25.81]. No events involving major SFP leakage have been reported. However, some events involved small leaks or potential leaks are reported. These events generally involved relatively small leak rates (less than about 50 gallons (about 0.19 m^3) per day). The latest EPRI's report for SFP PSA [Ref-25.82] introduces a value of 1.0E-6 /y for the initiating event frequency of "Loss of Spent Fuel Pool Inventory".

One of the intents of evaluating this initiator is to model the different nature of initiating events from loss of FPC, which means rapid water decrease by draindown. For this purpose, the order of magnitude of the EPRI report for the initiating event frequency is judged suitable to the initiator of loss of SFP liner in UK ABWR SFP PSA.

Siphoning of Spent Fuel Pool Inventory/ SFP draindown

Since the SFP pressure is comparatively low, piping break is very unlikely as a cause of siphoning. But siphoning itself is not precluded due to large uncertainty. Leakage from the valves rather than pipe break would dominate the cause. The impact of this event would be the SFP draindown from the normal water level to the bottom of SFP.

FPC has injection lines which potentially cause water decrease due to the siphon from the following failures:

- Siphon Breaker failure
- Check valve internal leak
- Manual valves / check valve external leak

The FPC injection line includes both manual valves and check valves. The frequency of this IE is calculated by failure rate of external leak from the manual isolation valves.

Spent Fuel Pool Gate / Seal failure

The SFP Gate provides the sealant barrier for the SFP. Gate or its seal failure would allow leakage into the reactor well and could result in partial draining of the SFP water. Loss of SFP inventory is considered to occur mainly by induced external force such as seismic induced failure. However, failure of the gate/seal by static load is modelled in order to capture the importance of this event in the SFP PSA as well as liner failure.

The IE frequency assigned is the same as that for Loss of Spent Fuel Pool Liner (see above).

Summary of IE Frequencies Quantification

Table 25.9.2-1 summarises the initiating event frequencies for UK ABWR SFP PSA. Summary of the initiating event frequencies for each POSs is described in PSA Summary report section 25.9 [Ref-25.1].

Initiating event frequencies for support system initiators are also considered to be applied to outage conditions even though some field data is obtained from study for normal operation. This is because reliability of a single support system division is considered not significantly different due to the maintenance conditions in other divisions of same system.

Table 25.9.2-1 Summary of SFP Initiating Event Frequencies for UK ABWR SFP PSA

<table>
<tr><th>No</th><th colspan="3">Initiating Event Group</th><th colspan="2">Frequency [/y][*1]</th><th>Related Area</th></tr>
<tr><td>1-1</td><td colspan="2" rowspan="2">Loss of SFP cooling</td><td>Loss of FPC</td><td colspan="2">-[*2]</td><td>IE analysis for SFP</td></tr>
<tr><td>1-2</td><td>Loss of RHR (POS E)</td><td colspan="2">-[*2]</td><td>IE analysis for SFP</td></tr>
<tr><td rowspan="4">2</td><td colspan="3" rowspan="4">Loss of Offsite Power</td><td>Plant centred</td><td>2.00E-03</td><td rowspan="4">Section 25.9.2.2
IE analysis for SFP</td></tr>
<tr><td>Switchyard centred</td><td>1.40E-02</td></tr>
<tr><td>Grid related</td><td>1.67E-02</td></tr>
<tr><td>Weather related</td><td>5.00E-03</td></tr>
<tr><td>3</td><td colspan="2" rowspan="3">Reactor Event Impacting the SFP</td><td>LOCA (LOCA outside containment + ISLOCA)</td><td colspan="2">6.98E-06 (POS F)</td><td>IEAP Level 1 PSA</td></tr>
<tr><td>4</td><td>Containment failure followed by ATWS</td><td colspan="2" rowspan="2">Each containment failure mode frequency × conditional probability</td><td rowspan="2"></td></tr>
<tr><td>5</td><td>Containment failure followed by failure of long term heat removal from containment</td></tr>
<tr><td>6</td><td rowspan="7">Support system initiator</td><td rowspan="4">Loss of SFP cooling</td><td>Loss of RCW/RSW (SFP)</td><td colspan="2">-[*2]</td><td rowspan="7">IE analysis for SFP</td></tr>
<tr><td>7</td><td>Loss of HVAC-R/BEEE/Z HVAC</td><td colspan="2">-[*2]</td></tr>
<tr><td rowspan="2">8</td><td>Loss of failure of HVAC-Hx/B</td><td colspan="2">-[*2]</td></tr>
<tr><td>Loss of Class 1 AC</td><td colspan="2">-[*2]</td></tr>
<tr><td>9</td><td rowspan="3">LOOP</td><td>Loss of HVAC-TBNEE/Z</td><td colspan="2">-[*2]</td></tr>
<tr><td>10</td><td>Loss of Class 3 AC</td><td colspan="2">-[*2]</td></tr>
<tr><td>11</td><td>Loss of HNCW</td><td colspan="2">-[*2]</td></tr>
<tr><td>12</td><td colspan="2" rowspan="6">Loss of SFP inventory</td><td>Failure associated with RHR connections to SFP</td><td colspan="2">-</td><td></td></tr>
<tr><td>13</td><td>Equipment / crane falling into the SFP</td><td colspan="2" rowspan="2">Small leak[*3]
1.43E-08 (POS F)
4.69E-09(POS A,C)</td><td rowspan="2"></td></tr>
<tr><td>14</td><td>Dropped cask/heavy load drop</td></tr>
<tr><td>15</td><td>Loss of Spent Fuel Pool Liner</td><td colspan="2">1.0E-6</td><td>EPRI document [Ref-25.82]</td></tr>
<tr><td>16</td><td>Siphoning of Spent Fuel Pool Inventory / SFP draindown</td><td colspan="2">1.60E-15</td><td>IE analysis for SFP</td></tr>
<tr><td>17</td><td>Spent fuel pool Gate / Seal Failure</td><td colspan="2">1.0E-6</td><td>EPRI document [Ref-25.82]</td></tr>
</table>

[*1] Including POS-B frequency
[*2] This is calculated for each POSs. Frequency for each POSs described in PSA Summary report section 25.9 [Ref-25.1].
[*3] Large leak is deleted by assumption on heavy load drop.

25.9.3 Event Sequence Analysis

The Event Sequence Analysis for the SFP PSA follows the same basic procedure as that of the Internal Events At Power PSA, see Section 25.4 and 25.7.

25.9.3.1 Event Sequence Modelling

The basic procedure for event tree development was as follows:

(1) Identify initiating event groups based on similar challenges to safety functions

(2) Create basic structure of event tree from required safety functions

(3) Describe each sequence end point (success or failure)

(4) Determine Operator intervention

(5) Define Plant Damage States

(1) Initiating event grouping for event tree

Initiating event grouping was developed for the SFP PSA. This resulted in four categories: those IEs which cause a loss of SFP cooling without a loss of SFP integrity (directly or due to support system failure, LOOP, and due to reactor accident progression), and those IEs which cause a loss of SFP water inventory. These are discussed in more detail below.

<u>Loss of SFP Cooling</u>

Initiating events which cause an interruption of the normally operating FPC System pump unit are evaluated as a loss of SFP cooling. This includes FPC System support systems initiators which may impact one or both FPC System units. The specific impact on the FPC System is captured in the analysis of each initiating event.

<u>Loss of Offsite Power</u>

Interruption of the FPC System may also be caused by a loss of offsite power (LOOP) initiating event, by a conditional LOOP given occurrence of any initiating event. A LOOP trips the operating FPC System pump, but it can be restarted by the operator if onsite power sources are available. A separate event tree is used to capture the unique aspects of LOOP events, including functioning of onsite power sources, recovery of offsite power sources, and recovery of SFP cooling.

<u>Other Accident Sequences Impacting SFP Cooling</u>

Operation of the FPC System may also be impacted by the accident progression of reactor events, due to adverse environmental conditions or structural damage to the reactor building. Such sequences may be successful in that core damage is avoided, but result in failure of the SFP cooling function, and so require mitigation actions to restore or maintain SFP decay heat removal and inventory. Reactor accident sequences which involve loss of RCS inventory outside containment, containment failure, or venting may result in adverse environmental conditions in the reactor building, and are considered initiating events if they can result in failure of the FPC System.

Loss of SFP Inventory

Events which result in leakage from the SFP are evaluated as a loss of SFP water inventory. The event is assumed to be initiated by a heavy load drop resulting in breach of the SFP liner.

A loss of SFP inventory may also be caused by the accident progression of reactor events, due to structural damage to the reactor building if it is sufficiently severe to result in either a liner breach or a loss of structural integrity of the SFP. Such events would involve catastrophic failure of the containment structure due to accident progression (i.e., hydrogen detonations or other dynamic impacts) or due to a beyond-design basis seismic event.

A load falling into the SFP could land on irradiated fuel causing mechanical damage and a release of fission products to the SFP water. Fission products would be released to the environment through the reactor building. It is evaluated that the release is limited from the fuel failure and the impact on the environment is very small in comparison.

A drop of a single fuel assembly into the SFP may result in localised fuel damage to the dropped assembly and to any fuel assemblies which it strikes. Releases associated with such events would be small compared to global fuel damage events.

(2) Development of Spent Fuel Pool Accident Progression

The event tree structure for the SFP PSA represents the accident progression, and includes the initiating event, diagnosis of the event, and manual initiation and operation of mitigating systems which are needed to restore the safety function(s). The specific diagnosis and mitigating systems depends upon the initiating event group being evaluated. Where the specific initiating event has an impact on the mitigating systems, this is addressed in the fault tree logic for that system.

Diagnosis refers to operator identification that the initiating event (e.g., either loss of SFP cooling or loss of SFP inventory) has occurred, and determining which backup systems need to be initiated to mitigate the event. The operators receive various SFP level and temperature indications and alarms.

The evaluation of the diagnosis of the event is addressed in the fault tree logic, and includes dependency in initiating multiple backup systems. That is, if the operator fails to properly diagnose the need to establish decay heat removal or inventory make-up from the first available system, he is likely to make the same diagnosis error to establish decay heat removal or inventory make-up from a second or third backup system.

In the event tree structure, the required system functions are impacted by the success or failure of the preceding system functions, and this dependency directly impacts sequence progression. For example, if the initiating event is a LOOP, then successful emergency AC power operation or the early recovery of offsite power permits recovery of decay heat removal using the FPC and RHR systems, while a prolonged loss of power requires SFP make-up due to loss of SFP inventory due to heatup and boiling.

Availability of support systems required for successful operation of the front-line systems is considered in system fault trees.

(3) Sequence End point

The sequence end point is defined as either of the following:

- Safe and stable SFP conditions with no on-going degradation for a minimum of 24 hours (mission time)
- Fuel damage due to uncovery of irradiated fuel in the SFP

All the sequences are developed to the point they are safe and stable or are shown to lead to fuel uncovery and damage.

The mission time is used to define a safe and stable condition after an initiating event has occurred and mitigation actions have been taken. Generally, a 24-hour mission time is applied; it is assumed in the PSA that if safe and stable (non-degrading) SFP conditions are maintained for the mission time, then plant recovery and equipment repairs will be accomplished to return to the conditions existing prior to the initiating event while maintaining SFP conditions. This is a typical assumption used in PSAs, and is based on the availability of additional staff and offsite resources following an accident with successful mitigation.

Longer mission times may be required if scenarios are identified which involve establishing interim mitigation measures to prolong the onset of fuel damage until other mitigation measures which are effective in the long term can be established. If the long term mitigation measures are not available until beyond 24 hours, then a longer mission time is needed for the interim mitigation system(s).

For the SFP PSA, although the scenarios extend beyond 24 hours, no mitigation systems are assumed or required to be operating during this extended time, and so the 24-hour mission time after successful restoration of either the SFP Decay Heat Removal or SFP Inventory Make-up functions is appropriate for this analysis.

(4) Operator intervention

Operator intervention is considered for manual backup of failure of an automated system to actuate, or manual initiation for a system without automatic actuation. Human reliability is estimated for each operator intervention considering its task and, if applicable, and sequence-specific considerations.

Table 25.9.3-1 shows the operator intervention considered in the SFP PSA.

Table 25.9.3-1 Operators Intervention Considered in Specific Initiating Event/Condition

Initiating Event	Operator Intervention
All initiating events	Manual initiation of standby FPC unit Manual initiation of RHR in FPA Mode (during shutdown conditions only) Alternate Injection by MUWC Alternate injection by SPCU Alternate injection by FLSS Alternate injection by FLSR Alternate injection by FP
LOOP	Recovery of offsite power Manual restart of FPC pump after LOOP
Loss of SFP Inventory	Isolation of SFP liner leakage

(5) Plant Damage States

Fuel damage is conservatively assumed as fuel unconvery.

25.9.3.2 Event Tree Development

The following is a brief description of the event tree 'top events' for each IE group.

(1) Sequence Description - Loss of SFP Cooling Event Tree

(a) Initiating Event - Loss of SFP Cooling

The FPC System is the primary means for removing the decay heat from spent nuclear fuel. Failure of the running train of the system would result in interruption of cooling and SFP heatup.

In the POS E, spent fuel is cooled by RHR because of full core off load. Thus, this initiating event in POS E is considered as "Loss of RHR".

(b) Diagnosis of Loss of SFP Cooling (Temperature)

This top event (DIAG_FPC) addresses operator diagnosis that a loss of SFP cooling has occurred and that manual start of the standby FCP train is required. The initial loss of the FPC System running train would be the earliest and most direct indication. The SFP temperature instrumentation indications and alarms provide a subsequent cue for recovery if cooling is not restored. For this analysis, diagnosis is credited on loss of the SFP cooling running train, and it depends on subsequent temperature indications and alarms.

Success of this event permits consideration of restoration of SFP cooling by the standby FPC train or RHR in FPA mode.

Failure of this event results in SFP boiling and loss of inventory due to boiloff, and so requires SFP make-up.

(c) Standby FPC Train

This top event (FPC) addresses operation of the standby FPC train after the operator successfully diagnoses that the running FPC train is not in service. This event addresses both the operator action to align and start the train, as well as equipment failures. Some initiators which cause a loss of the running train also affect the standby train; these dependencies are captured in the system fault trees. In POS E, the FPC System is not credited.

Success of this event assures adequate decay heat removal and SFP inventory to prevent any release from the SFP.

Failure of this event would require consideration of RHR in FPA mode as an alternative for SFP cooling, if the system is available.

(d) Backup FPC – RHR in FPA Mode

This top event (BU_FPC) addresses alignment and operation of the RHR System in an alternate alignment for FPA Mode. This node is applicable if an RHR train is available while the reactor is shutdown, and if SFP inventory is maintained in the SFP. Therefore, RHR in FPA Mode must be initiated prior to SFP boiling and lowering of SFP level below the FPC System connection. Diagnosis failure, which accounts for the time available, is included under the supporting fault tree logic and uses the same HRA event as is used for the Loss of SFP Cooling initiating event for starting the standby FPC unit. RHR in FPA mode is not credited in POS F (at Power operation).

Success of this event assures adequate decay heat removal and SFP inventory to prevent any release from the SFP.

Failure of this event would result in SFP boiling and loss of inventory due to boiloff, and so require SFP make-up.

(e) Diagnosis of Loss of SFP Cooling (Level)

This top event (DIAG_FPB) addresses operator diagnosis that a loss of SFP cooling has occurred and progressed to the point of SFP boiling and loss of inventory, such that SFP make-up is required. The initial loss of the FPC System and failure to start and run of the standby train would be the earliest and most direct indication that make-up is required. The SFP level instrumentation indications and alarms provide a subsequent cue for recovery if cooling is not restored. For this analysis, no credit is taken for diagnosis by loss of the SFP cooling running train, and only subsequent level and temperature indications and alarms are evaluated. This event is assessed only after an initial failure to diagnose the loss of the running FPC train, crediting the additional cues provided by level indications and alarms as the SFP begins to boil.

Success of this event permits consideration of SFP make-up options to maintain SFP water level above the top of the irradiated fuel.

Failure of this event results in SFP water level lowering to the top of the irradiated fuel.

(f) SFP Make-up

This top event (SFP_MU) addresses alignment and manual initiation of the make-up options to the SFP which are credited for loss of inventory due to boiling. The make-up options are as follows:

- MUWC System
- SPCU System
- RHR System (if not POS F at Power operation, i.e., the reactor is shut down)
- Fire Protection System water injection

Success of this event assures maintenance of adequate SFP inventory above the top of the irradiated fuel using one of the above make-up systems and avoids of any release from the SFP.

Failure of this event would require consideration of the FLSS and FLSR make-up options.

(g) SFP Make-up by FLSS

This top event (FLSS) addresses alignment and operation of the make-up option of the FLSS which provides water inventory that maintains irradiated fuel covered in the SFP, thus preventing fuel damage. The FLSS has an alternate generator power supply which can be used for scenarios involving a loss of power.

Success of this event assures maintenance of adequate SFP inventory above the top of the irradiated fuel and avoids of any release from the SFP.

Failure of this event would require consideration of the FLSR make-up option.

(h) SFP Make-up by FLSR

This top event (FLSR) addresses alignment and operation of the make-up option of the FLSR which provides water inventory that maintain coverage of irradiated fuel in the SFP, thus preventing fuel damage.

Success of this event assures maintenance of adequate SFP inventory above the top of the irradiated fuel and avoids of any release from the SFP.

Failure of this event would result in uncovery of the irradiated fuel and assumed fuel damage.

(i) Loss of SFP Cooling Sequences Leading to Fuel Damage

For a Loss of SFP Cooling initiating event, three sequences lead to fuel damage:

Sequence SFPC6: After a Loss of SFP Cooling initiating event, the operator successfully diagnoses the event, but is unable to start the standby FPC train. RHR in FPA Mode fails or is unavailable during reactor operation. SFP Make-up, FLSS, and FLSR all fail to maintain SFP inventory. The result of this sequence is fuel damage.

Sequence SFPC10: After a Loss of SFP Cooling initiating event, the operator fails to diagnose the initial SFP loss of cooling; subsequent cues on SFP level allow recovery of this diagnosis failure, and the operator recognises the need to initiate make-up systems to accommodate boiloff due to decay heat. The sequence is similar to SFPC6, and SFP Make-up, FLSS, and FLSR all fail to maintain SFP inventory. The result of this sequence is fuel damage.

Sequence SFPC11: After a Loss of SFP Cooling initiating event, the operator fails to diagnose the initial SFP loss of cooling. Subsequent cues on SFP level would allow recovery of this diagnosis failure, but this is not successful and not recovered when SFP water level lowers below the top of the irradiated fuel in the SFP. The result of this sequence is fuel damage.

(j) Loss of SFP Cooling due to reactor challenge

In this SFP PSA, reactor challenges which also have an effect on SFP are also considered as initiating events. When ISLOCA or BOC occurs in reactor building, each room in the reactor building including FPC pump room is assumed to be steamy causing “Loss of SFP Cooling”. When containment failure occurs, leaked steam from the containment may go into the reactor building, and cause “Loss of SFP Cooling”. The resulting event sequence is similar to the above discussion. However, SSCs in the reactor building are not credited.

(2) Sequence Description - Loss of Offsite Power Event Tree

(a) Initiating Event - LOOP

A LOOP, as a direct initiating event or as a conditional event after a plant scram, would interrupt AC power to the FPC System operating pump and cause a loss of SFP cooling. This event differs from other loss of SFP cooling initiating events since the FPC system is undamaged and could be restored to operation.

Treatment of Consequential LOOP in SFP

The consequential LOOP does not occur by the SFP initiating event.

On the other hand, the consequential LOOP sequences by these reactor challenges are considered. In this case, the event sequence for the reactor challenge is linked to the SFP event tree as an initiating event and if it includes LOOP condition, the reactor challenge is also considered in each mitigation system as a dependent failure.

(b) Emergency Diesel Generators

This top event (EDG) addresses automatic start and operation of emergency power prior to either boiling or fuel uncovery. Successful operation of emergency power permits recovery of SFP cooling either by the FPC System or by the RHR System in FPA Mode, either of which is powered by the emergency diesel generators systems (EDGs). If SFP cooling is not restored despite the availability of the emergency power, some systems needed for SFP make-up can be powered by the EDGs.

Success of this event allows consideration of recovery of SFP cooling.

Failure of this event would require consideration of the recovery of offsite power.

(c) Recovery of Offsite Power (Early)

This top event (OSP_EARLY) addresses recovery of offsite power prior to SFP boiling. If the EDGs fail to operate, SFP cooling can still be recovered if offsite power is recovered before SFP boiling begins.

Success of this event allows consideration of recovery of SFP cooling.

Failure of this event is assumed to result in SFP heatup and boiling, with subsequent loss of SFP inventory due to boiloff, and would require consideration of late recovery of offsite power.

(d) SFP Cooling

Success of this event assures adequate decay heat removal and SFP inventory to prevent any release from the SFP.

Failure of this event would require consideration of backup fuel pool cooling by the RHR System in FPA Mode.

(e) Backup FPC – RHR in FPA Mode

Success of this event assures adequate decay heat removal and SFP inventory to prevent any release from the SFP.

Failure of this event would result in SFP boiling and loss of inventory due to boiloff, and so require SFP make-up.

(f) Recovery of Offsite Power (Late)

This top event (OSP_LATE) addresses recovery of offsite power prior to irradiated fuel uncovery. If the EDGs fail to operate and offsite power is not recovered before SFP boiling occurs, SFP make-up can still be recovered if offsite power is recovered before SFP boiling results in core uncovery. Recovery of offsite power prior to fuel uncovery is modelled in order to properly model SFP make-up, since these systems require either offsite or emergency AC power.

Success of this event allows consideration of SFP make-up to maintain SFP inventory above the top of the irradiated fuel.

Failure of this event would only permit consideration of SFP make-up from FLSS or FLSR, which are able to function independent of offsite or emergency AC power.

(g) Diagnosis of Loss of SFP Cooling (Level)

Success of this event permits consideration of available (depending upon the status of the emergency and offsite power sources) SFP make-up options to maintain SFP water level above the top of the irradiated fuel.

Failure of this event results in uncovery of the irradiated fuel and assumed fuel damage.

(h) SFP Make-up

Success of this event, with successful emergency power or offsite power recovery, assures maintenance of adequate SFP inventory above the top of the irradiated fuel and avoids of any release from the SFP.

Failure of this event would require consideration of the FLSS and FLSR make-up options.

(i) SFP Make-up by FLSS

The FLSS includes an alternate generator power supply which can function in the event of a LOOP.

Success of this event assures maintenance of adequate SFP inventory above the top of the irradiated fuel and avoids of any release from the SFP.

Failure of this event would require consideration of the FLSR make-up option.

(j) SFP Make-up by FLSR

Success of this event assures maintenance of adequate SFP inventory above the top of the irradiated fuel and avoids of any release from the SFP.

Failure of this event would result in uncovery of the irradiated fuel and assumed fuel damage.

(k) Loss of Offsite Power Sequences Leading to Fuel Damage

For a LOOP initiating event, eight sequences lead to fuel damage:

Sequence SFPLOOP6: After a loss of offsite power, the EDGs successfully start and operate, but SFP cooling is not recovered. RHR in FPA Mode fails to provide SFP cooling, resulting in SFP heatup and boiling. The operator successfully diagnoses the SFP loss of cooling and the need to initiate make-up systems to accommodate boiloff due to decay heat. The SFP Make-up, FLSS and FLSR all fail to maintain spent fuel pool inventory. The result of this sequence is fuel uncovery and damage.

Sequence SFPLOOP7: This sequence is similar to SFPLOOP6 except the operator fails to diagnose the SFP loss of cooling and the need to initiate make-up systems to accommodate boiloff due to decay heat, and this is not recovered when SFP water level lowers to below the top of the irradiated fuel in the SFP. The result of this sequence is fuel uncovery and damage.

Sequence SFPLOOP13: After a loss of offsite power, the EDGs fail to operate, but offsite power is recovered before SFP boiling occurs. SFP cooling by the FPC System and RHR in FPA Mode both fail resulting in SFP heatup and boiling. The operator successfully diagnoses the SFP loss of cooling and the need to initiate make-up systems to accommodate boiloff due to decay heat. SFP Make-up, FLSS and FLSR all fail to maintain spent fuel pool inventory. The result of this sequence is fuel uncovery and damage.

Sequence SFPLOOP14: This sequence is similar to SFPLOOP13 except the operator fails to diagnose the SFP loss of cooling and the need to initiate make-up systems to accommodate boiloff due to decay heat, and this is not recovered when SFP water level lowers to below the top of the irradiated fuel in the SFP. The result of this sequence is fuel uncovery and damage.

Sequence SFPLOOP18: After a loss of offsite power, the EDGs fail to operate followed by failure to recover offsite power before SFP boiling occurs. SFP cooling by the FPC System and RHR in FPA Mode are not available due to the loss of power, resulting in SFP heatup and boiling. Offsite power is recovered prior to fuel uncovery, and the operator successfully diagnoses the SFP loss of cooling and the need to initiate make-up systems to accommodate boiloff due to decay heat. SFP Make-up, FLSS and FLSR all fail to maintain spent fuel pool inventory. The result of this sequence is fuel uncovery and damage.

Sequence SFPLOOP19: This sequence is similar to SFPLOOP18 except the operator fails to diagnose the SFP loss of cooling and the need to initiate make-up systems to accommodate boiloff due to decay heat, and this is not recovered when SFP water level lowers to below the top of the irradiated fuel in the SFP. The result of this sequence is fuel uncovery and damage.

Sequence SFPLOOP22: After a loss of offsite power, the EDGs fail to operate followed by failure to recover offsite power before SFP boiling occurs. SFP cooling by the FPC System and RHR in FPA

Mode are not available due to the loss of power, resulting in SFP heatup and boiling. Offsite power is not recovered prior to fuel uncovery, but the operator successfully diagnoses the SFP loss of cooling and the need to initiate make-up systems to accommodate boiloff due to decay heat. SFP Make-up is not available due to the loss of power, and FLSS and FLSR all fail to maintain spent fuel pool inventory. The result of this sequence is fuel uncovery and damage.

Sequence SFPLOOP23: This sequence is similar to SFPLOOP22 except the operator fails to diagnose the SFP loss of cooling and the need to initiate make-up systems to accommodate boiloff due to decay heat, and this is not recovered when SFP water level lowers to below the top of the irradiated fuel in the SFP. The result of this sequence is fuel uncovery and damage.

(3) Sequence Description - Loss of SFP Inventory

(a) Initiating Event - Loss of SFP Inventory

SFP liner breach or draindown represents a loss of SFP inventory. It has previously been identified as the dominant contributor to SFP-related risks. It can be related to any of the following:

(i) Direct failure of the SFP liner due to a heavy load drop either directly into the SFP or to the reactor building floor

(ii) Failure of the SFP gates or seals when at power, causing a partial draindown to the bottom of the transfer canal

(iii) Failure of the crane or heavy load (e.g., cask drop) into the SFP or its gates, causing physical rupture similar to (i) or (ii) — this can be a random event or a seismic-induced event.

(iv) Siphoning of the SFP due to a break in a line that extends down below the surface of the SFP. Typical suction and discharge lines on SFPs do not extend far down into the pool or are equipped with anti-siphon devices that protect against the possibility of siphoning the pool. For the UK ABWR, no penetrations are present below SFP normal water level except the SFP gates. Siphoning events are quantified as failure of the anti-siphon device and loss of inventory.

(b) SFP Structural Integrity

This top event (SFP_SI) addresses the structural integrity of the SFP, that is, the concrete supporting structure. If this is damaged severely, the SFP may lose integrity and a catastrophic loss of inventory (greater than available make-up) could occur. This may occur if a heavy load is dropped onto the edge of the SFP.

Success of this event would permit consideration of leak isolation or make-up to the SFP.

Failure of this event is assumed to result in uncovery of the irradiated fuel and assumed fuel damage.

(c) SFP Automatic Make-up

This top event (SFP_MU_A) addresses automatic make-up to the SFP skimmer by the MUWC system. If the SFP pump is still operating, then this automatic make-up function is available to maintain SFP inventory for a liner breach.

Success of this event assures maintenance of adequate SFP inventory to permit continued operation of the FPC system, which assures adequate decay heat removal to prevent any release from the SFP.

Failure of this event is assumed to require operator intervention to diagnose and isolate the leak or maintain SFP inventory by manual make-up.

(d) Diagnosis of Loss of SFP Inventory

This top event (DIAG_INV) addresses operator diagnosis that a loss of SFP inventory has occurred and that SFP make-up for is required. The SFP level instrumentation indications and alarms provide a cue that action is required. As level continues to lower, the FPC System will fail due to SFP level lowering below the pump intake connection, and additional alarms as well as indication of increasing SFP temperature are provided.

Success of this event would permit consideration of leak isolation or make-up to the SFP.

Failure of this event results in SFP water level lowering to the top of the irradiated fuel.

(e) SFP Leak Isolation

This top event (SFP_ISOL) addresses isolating a SFP liner leak. There is the potential of isolating some leaks where structural integrity of the SFP is maintained. Since loss of SFP inventory due to a leak is caused by the liner failure, all leak flow goes through the leak detection lines. The operator can respond to the alarm and isolate the line.

Success of this event assures maintenance of adequate SFP inventory above the top of the irradiated fuel, and avoids of any release from the SFP. With the leak isolated, SFP heat removal can either continue uninterrupted or, if necessary, the SFP can be refilled to permit restoration of normal heat removal. It is assumed that once the leak is successfully diagnosed and isolated, that restoration of normal SFP water level and heat removal occurs, and additional failures are not assessed.

Failure of this event would require SFP make-up.

(f) SFP Make-up

Success of this event assures maintenance of adequate SFP inventory above the top of the irradiated fuel and avoids of any release from the SFP.

Failure of this event would require consideration of the FLSS and FLSR make-up options.

(g) SFP Make-up by FLSS

Success of this event assures maintenance of adequate SFP inventory above the top of the irradiated fuel and avoids of any release from the SFP.

Failure of this event would require consideration of the FLSR make-up option.

(h) SFP Make-up by FLSR

Success of this event assures maintenance of adequate SFP inventory above the top of the irradiated fuel and avoids of any release from the SFP.

Failure of this event would result in uncovery of the irradiated fuel and assumed fuel damage.

(i) Loss of SFP Inventory Due to Liner Failure Sequences Leading to Fuel Damage

For a Loss of SFP Inventory Due to Liner Failure event, three sequences lead to fuel damage:

Sequence SFPIL6: After a SFP loss of inventory event which does not cause failure of the structural integrity of the SFP, automatic make-up to the SFP skimmer fails, but the operator successfully diagnoses the event. Isolation of the leak fails, and the operator attempts to maintain SFP inventory. SFP Make-up, FLSS, and FLSR all fail to maintain SFP inventory. The result of this sequence is fuel uncovery and damage.

Sequence SFPIL7: This sequence is similar to SFPIL5 except the operator fails to diagnose the SFP loss of inventory and the need to initiate make-up systems to maintain SFP level, and this is not recovered when SFP water level lowers to below the top of the irradiated fuel in the SFP. The result of this sequence is fuel uncovery and damage.

Sequence SFPIL8: A loss of SFP inventory occurs and involves a SFP structural integrity failure. This is assumed to result in catastrophic loss of SFP inventory such that make-up to the SFP is inadequate to maintain level. The result of this sequence is fuel uncovery and damage.

(j) Loss of SFP Inventory Due to reactor challenge

In this SFP PSA, reactor challenges which also affect the SFP are also considered as initiating event. When the containment failure occurs and if the containment stress/strain effect on SFP structure or hydrogen burning occurs because of leaked hydrogen from the containment, these effects also cause “Loss of SFP Inventory”. These are also considered as “Loss of SFP Inventory” and its event sequence is similar to the above discussion. However, certain SSCs are not credited in this event.

(4) Summary of credited system condition in each POS

Table 25.9.3-2 describes unavailable condition of systems for each POS. Event trees for each POS are developed considering the mitigation systems described in Table 25.9.3-2.

The success criteria for each initiating event in SFP PSA are summarised in Table 25.9.3-3. In this success criteria analysis, different criterion should be applied to each POS. Thus, POS specific event trees and functional fault trees are developed in SFP PSA model.

25.9.3.3 Determination of Success Criteria

(1) Definition of Safety Functions

Critical safety functions considered in the at Power Level 1 PSA are as follows.

- Reactivity Control
- Core Cooling
- Containment Heat Removal
- Reactor Coolant Pressure Boundary Protection
- Vapour Suppression
- Containment Isolation

Applicability of each safety function to each POS in SFP PSA is discussed in the following.

<u>Reactivity Control</u>

The SFP PSA begins with a condition in which reactivity control using the spent fuel storage rack has already been established.

The potential IEs categorised as the broad category "reactivity insertion" are screened from the SFP PSA scope (e.g., calculating the scenarios leading to global fuel damage). Therefore, the reactivity control function is not modelled in the SFP PSA. Reactivity insertion potentially leads to localised fuel heat up and damage is addressed as the Target 8 assessment by referencing the deterministic consequence evaluation.

<u>Fuel Cooling</u>

Since the SFP PSA treats the fuel in SFP, core cooling function is called the fuel cooling function in the SFP PSA. Fuel cooling must be maintained for all events, and therefore considered in the SFP PSA.

There are two safety functions for fuel cooling:

- Decay heat removal through heat exchanger(s)
- Makeup of SFP

The decay heat removal function transfers the decay heat using a closed loop which consists of pump(s) and heat exchanger(s). FPC and RHR provide this function. This function succeeds if a system(s) is initiated before unacceptable conditions are reached, including loss of suction water. Since the decay heat removal function will not compensate for the loss of coolant inventory, this function is not used for mitigating a loss of SFP inventory events.

The makeup function is to provide sufficient amount of water to SFP to compensate the boil-off or leakage, using water injection systems such as MUWC or FLSS.

<u>Containment Heat Removal</u>

In terms of SFP cooling, the containment heat removal function is not required. The decay heat is removed through the heat exchanger(s) or by releasing steam (during evaporation) through the refuelling deck in the reactor building.

Reactor Coolant Pressure Boundary Protection

In terms of SFP PSA, the RCPB function is not relevant to the SFP PSA.

Vapour Suppression

In terms of SFP PSA, the vapor suppression is not relevant to the SFP PSA.

Containment Isolation

In terms of SFP PSA, the containment isolation is not relevant to the SFP PSA.

Structural Integrity

Structural integrity is considered in the SFP PSA.

Structural integrity consists of the concrete and the leak tightness of the SFP liner. In terms of SFP PSA, structural integrity is required to maintain the appropriate water level in SFP.

Reactor Building Hydrogen Control

Reactor Building hydrogen control may be critical to ensure the continued integrity of the makeup pathways to the SFP and the integrity of the SFP structure. When the hydrogen is generated from SFP, it is controlled and removed by Standby Gas Treatment System (SGTS), Passive Auto-catalytic Recombiner (PAR) and a blowout panel. In this situation, the fuel in the SFP is already damaged. Therefore, this situation is treated in the Level 2 SFP PSA. In the Level 2 SFP PSA, the availability of above hydrogen control systems is discussed.

Summary of Safety Functions Considered in Level 1 SFP PSA

In summary, the following safety functions are considered in the Level 1 SFP PSA for each POS.

- Fuel cooling consists of decay heat removal and makeup
- Structural Integrity consists of concrete and leak tightness of the liner

Treatment of SFP overfill

Manual injection systems can be diagnosed by operator, requiring the operator to stop the makeup function before overfill of the SFP. This failure probability of diagnosis is negligible. On the other hand, recognizing the failure of the automatic stop signal (MUWC) is more difficult for the operator to diagnose. In this case, the SFP is overfilled and the water is drained or funnel into a building equipment shaft or stairs room. However, the flow rate of the MUWC (automatic injection) is 32 m^3/hr. If the MUWC continues to inject after overfill for 24 hr, its total water volume is 768 m^3. This water volume is very low compared to the 3,600 m^3(S/P water volume) of flooding criteria [Ref-25.83]. Thus, the SFP overfill is not considered in the SFP PSA.

(2) Definitions of Fuel Damage

The definition of global fuel damage is the onset of fuel uncovery (i.e. collapsed water level decreases below the Top of Active Fuel (TAF) in the SFP).

Fuel Cooling

Failure of the fuel cooling function would cause global fuel damage. Therefore, the acceptance criterion is to keep the collapsed water level above the TAF in the SFP.

Structural Integrity

Loss of structural integrity would cause a SFP water (inventory) decrease and lead to fuel damage due to uncovery. Therefore, the acceptance criterion is to maintain the SFP water level including isolation of leak detection line before fuel uncovery.

(3) Establishment of System Success Criteria

This section establishes system success criteria for the SFP PSA.

Fuel cooling success criteria and associated elements are discussed in this section (3). Success criteria for use of mitigation functions are discussed in sections (4) to (6).

(a) Evaluation of Decay Heat

Decay heat evaluation is needed in order to:

- Identify the makeup system(s) which can remove decay heat.
- Evaluate the time to a particular relevant condition (e.g. boiling, fuel uncovery).

Decay heat from stored fuel is calculated using the May-Witt equation. The May-Witt equation provides conservative decay heat value compared to the detailed calculation by ORIGEN code. Use of the May-Witt equation for calculating decay heat of the spent fuels in the SFP is therefore considered to be reasonable.

Two cases are considered as the maximum heat load during non-full core offload (POS S, A, C, D, F) and full core offload (POS E).

Case 1: Normal heat Load

The Normal heat load means representative decay heat levels except for the full core off-load condition. The load for this case is the decay heat generated from the spent fuel in the SFP immediately after closing the SFP Gate when normal fuel replacement with the equilibrium core is performed. The heat load is the sum of the decay heat released by the extracted spent fuels with various cooling time.

Case 2: Maximum heat load

The Maximum heat load means the decay heat level of full core off load. The load for this case is the decay heat generated from the spent fuel in the SFP immediately after closing the SFP Gate when the whole fuel load in the core (after completion of an equilibrium cycle) is extracted and stored in the SFP. The heat load is the sum of the decay heat released by the extracted spent fuels with various burnup and cooling time.

The uncertainties (potential conservatism) associated with the decay heat and evaporation rate are as follows:

- The decay heat is calculated using the May-Witt equation.
- The decay heat POS is evaluated as representative of the maximum decay heat.
- An infinite time is conservatively used as the “irradiation period”.

These potential conservatisms are acceptable for the following arguments:

- One FPC division is sufficient for the highest decay heat against the evaluated conservative decay heat. Therefore, the current criteria will not change.
- One RHR division is sufficient for the decay heat in the full core off load against the evaluated conservative decay heat. Therefore, the current criteria will not change.
- One pump of any credited makeup systems is sufficient for the highest decay heat with design leak rate against the evaluated conservative decay heat. Therefore, the current success criteria will not change.

(b) Evaluation of Leak Rate

<u>Small Leak</u>

Opening of all welding lines on the bottom of the SFP is conservatively assumed. There are four leak detection lines under welding lines on the bottom of SFP to detect the leakage from SFP liner due to such as cracking. To calculate the small leakage from the SFP, opens of the whole welding line on the bottom of SFP is assumed conservatively.

<u>Large Leak</u>

SFP water level is assumed to decrease rapidly in the Loss of SFP inventory (Large leak). Leak rate is not specified other than leak beyond design leak rate. It is assumed that the large leak beyond the design leak rate is impossible to makeup, similar to RPV rupture. For leak beyond design leakage no credit is taken for mitigation functions. Therefore, time margin analysis did not need to be performed in the large leak condition.

(c) Treatment of Diagnosis error

In the Tabular Task Analysis (TTA) on HRA, human errors are regarded as tasks with cue by alarm/indication and procedure. Diagnosis errors were determined for the following:

- Diagnosis for SFP cooling (Functional FT ID: DIAG_FPC) (including backup and alternative SFP cooling action): This is cued by “SFP temperature high” alarm and procedure.
- Diagnosis for SFP injection (DIAG_FPB and DIAG_INV) (including alternative injection action): This is cued by “SFP water level low” alarm and procedure.

(4) Definition of Success Criteria against Loss of SFP Cooling

In this step, the system level success criteria for the cases of loss of SFP cooling events are discussed.

A 24 hour mission time is applied to the credited systems.

(a) Comparison to existing BWRs

The reasonableness and acceptability of the success criteria are checked by comparing to existing BWR: "Spent Fuel Pool Risk Assessment Integration Framework (Mark I and II BWRs) and Pilot Plant Application" [Ref-25.82]

The following success criteria (supported by analyses) are basically the same as EPRI's report.

- One RHR Fuel Pool Cooling Function
- FP as makeup water
- One FPC for normal heat load
- MUWC as makeup water

The following success criteria in EPRI's report are not used in the UK ABWR SFP PSA.

- Demineralised water via supply boxes: UKABWR does not have this function
- HPSW: UKABWR does not have this system.

The following success criteria, not credited in the EPRI report are also applied to for the UK ABWR. These criteria are realistic based on the UK ABWR design.

- SPCU is credited for makeup of SFP.
- FLSS is credited for makeup of SFP.
- FLSR is credited for makeup of SFP.
- Automatic makeup to the skimmer surge tank by MUWC is credited.

These investigations are applicable to Loss of SFP cooling, Loss of offsite power and Loss of SFP inventory.

(b) Systems for Decay heat removal function

<u>RHR</u>

The primary measure for mitigating the loss of decay heat removal type events (including LOOP) is to recover the decay heat removal function by starting or restarting RHR. One division of RHR Fuel Pool Cooling Function is capable of removing decay heat.

NOTE: RHRs have function of SFP cooling by using FPC assist mode. However, RHRs cannot be credited because RHRs are used or on standby to cope with reactor initiating event in reactor at power condition. Thus, RHRs are not credited in POS F.

<u>FPC</u>

FPC of UK ABWR has two independent divisions [Ref-25.84]. One division of FPC is capable of removing decay heat only for the normal heat load. For the maximum (full core offload), FPC cannot be credited (POS E). Therefore, FPC can be credited as the heat removal measure in the All POSs for SFP PSA except POS E.

(c) Systems for Makeup function

Even after loss of all decay heat removal measures (as both IE and mitigation system failures), the SFP water level can be maintained above TAF if sufficient makeup flow rate exceeding the evaporation rate is provided.

MUWC

Automatic makeup function to SFP skimmer surge tank: MUWC automatically supplies water to the SFP skimmer surge tank from the CST to compensate evaporation, leakage from liner cracks or overflow [Ref-25.84]. If the FPC keeps its normal operation mode (e.g. taking suction from the skimmer surge tanks and return water to SFP), MUWC essentially makes up the SFP (and reactor well during pool gate open).

The makeup flow to the SFP skimmer surge tank by MUWC is 32 m^3/h (assumed based on J-ABWR design and to be adopted by UK ABWR design) which exceeds the representative evaporation rates in all POSs for SFP PSA. Therefore, continuous operation of automatic makeup function for SFP skimmer surge tank by MUWC (combined with normally operating FPC) is credited as one of the success paths in all POSs except for POS E.

Manual makeup function to RPV or SFP: MUWC has injection path to SFP via RHR and FPC lines. However, MUWC is not defined as Class 1 or Class 2 makeup systems for SFP in the fault schedules in DBA [Ref-25.85].

The capacity of one MUWC pump is 125 m^3/h exceeds the evaporation rate in all the POSs for SFP PSA [Chapter 16 of PCSR Rev. C].

Since there is no automatic initiation function of the MUWC for RPV/SFP/reactor well makeup, only manual operation is considered for all the POSs.

SPCU

SPCU has SFP Makeup Mode via FPC. However, SPCU is not defined as Class 1 or Class 2 makeup systems for SFP makeup in the fault schedule [Ref-25.85].

The capacity of one SPCU pump is 250 m^3/h [Chapter 16 of PCSR Rev. C] exceeds the evaporation rate in all POSs.

Since there is no automatic initiation function of the SPCU, only manual operation is considered for all the POSs.

Fire Protection

The Fire Protection system has a SFP Makeup Mode via MUWC. However, the Fire Protection system is not defined as Class 1 or Class 2 makeup systems for SFP makeup in the fault schedule.

It is assumed that one Fire Protection system pump injection flow rate which exceeds the evaporation rate in all POSs.

Since there is no automatic initiation function of the Fire Protection system, only manual operation is considered for all POSs. In addition, it is assumed that the Fire Protection system does not need to use AC power supply because FP system has diesel driven pump.

FLSS

The design specification of SFP makeup flow rate is 120 m^3/h (at 0 MPa [gauge]), respectively, which exceeds the evaporation rates in all the POSs for SFP PSA. Operator action for FLSS operation is credited for all POSs for SFP PSA. FLSS has SFP spray function. This function is not credited in level 1 SFP PSA. The spray is credited in Level 2 PSA to prevent fuel heat up after the fuel is uncovered.

FLSR

FLSR is defined as one of the RPV, reactor well and SFP makeup systems [Ref-25.24]. Since FLSR requires preparation of mobile equipment, it is credited in the all POSs for SFP PSA.

The design specification of SFP makeup flow rate is 120 m^3/h (at 3.43 kPa [gauge]), respectively, which exceeds the evaporation rates in all the POSs.

Since there is no automatic initiation function of FLSR, only manual operation is considered.

(5) Definition of Success Criteria against Loss of Offsite Power

(a) Decay Heat Removal

For the front line function (Decay heat removal), the success criteria against loss of offsite power are the same criteria as that against loss of SFP cooling except the EDG is also required.

(b) Makeup Systems

For the front line function (Makeup system), the success criteria against loss of offsite power are the same criteria as that against loss of SFP cooling except the EDG is also required.

(c) Power Supply

AC power for FPC, MUWC, SPCU and RHR is normally supplied by offsite power. When LOOP occurred, EDG can supply AC power to FPC, MUWC, SPCU and RHR. In addition, BBG can be credited for AC power supply to FLSS even if EDG is lost. Moreover, FLSR and Fire Protection system does not need to use AC power supply because FLSR is mobile system and Fire Protection system has diesel driven pump (Assumption).

(6) Definition of Success Criteria against Loss of SFP Inventory

(a) Structural Integrity

The loss of structural integrity is caused by heavy load drop.

(b) Isolation Leakage

After loss of SFP inventory due to SFP liner failure, the leakage is detected by Fuel Pool Liner Drain Leakage Detection System. Operator can isolate the line and stop the leakage. Since MUWC can automatically inject sufficient water, the success criteria of this manual isolation do not need to define the time margin when MUWC succeeds.

(c) Makeup Systems

For the front line function (Makeup system), the success criteria against loss of SFP inventory are the same criteria as that against loss of SFP cooling except time margin.

If the automatic makeup fails, all makeup systems have to inject makeup water more than 35.6 m^3/h (NHL), which includes evaporated flow rate and small leak rate. In this case, all makeup functions can inject sufficient flow rate when they are manually initiated. The time margin is shorter than the Loss of SFP cooling scenario because of the difference of water loss rate. This time margin is considered in HRA.

Table 25.9.3-2 System Unavailability

POS \ System	FPC		RHR		MUWC			Fire Protection	SPCU	FLSS		FLSR	
	A	B	A	B	A	B	C			A	B	A	B
S	-	-	-	-	-	-	-	-	X	-	-	-	-
A	-	X	-	X	-	X	-	-	X	-	-	-	-
C	X	-	X	-	X	-	X	-	X	-	-	-	-
D	-	-	-	-	-	-	-	-	X	-*	-*	-	-
E	X	X	X	-	X	-	X	-	X	-	-	-	-
F	-	-	-	-	-	-	-	-	-	-	-	-	-

X: Unavailable, -: available

*: The SSCs in each division is alternately available in each POSs

Table 25.9.3-3 Summary of Success Criteria

Initiating events	POS	Systems							
Loss of SFP Cooling[4]	S	(1)RHR-B	(2)FPC-B	(3)MUWC-A,B,C		(5)FLSS-A,B	(6) FLSR-A,B	(7) Fire Protection	
	A			(3)MUWC-A,C		(5)FLSS-A,B	(6) FLSR-A,B	(7) Fire Protection	
	C			(3)MUWC-B		(5)FLSS-A,B	(6) FLSR-A,B	(7) Fire Protection	
	D	(1)RHR-A	(2)FPC-A	(3)MUWC-A,B,C		(5)FLSS-A	(6) FLSR-A,B	(7) Fire Protection	
	E			(3)MUWC-B		(5)FLSS-A,B	(6) FLSR-A,B	(7) Fire Protection	
	F		(2)FPC-A or B	(3)MUWC-A,B,C	(4)SPCU	(5)FLSS-A,B	(6) FLSR-A,B	(7) Fire Protection	
Loss of Offsite Power[1]	S	(1)RHR-B	(2)FPC-A,B	(3)MUWC-A,B,C		(5)FLSS-A,B	(6) FLSR-A,B	(7) Fire Protection	
	A		(2)FPC-A	(3)MUWC-A,C		(5)FLSS-A,B	(6) FLSR-A,B	(7) Fire Protection	
	C		(2)FPC-B	(3)MUWC-B,		(5)FLSS-A,B	(6) FLSR-A,B	(7) Fire Protection	
	D	(1)RHR-A	(2)FPC-A,B	(3)MUWC-A,B,C		(5)FLSS-A	(6) FLSR-A,B	(7) Fire Protection	
	E		(2)FPC-A	(3)MUWC-B		(5)FLSS-A,B	(6) FLSR-A,B	(7) Fire Protection	
	F		(2)FPC-A,B	(3)MUWC-A,B,C	(4)SPCU	(5)FLSS-A,B	(6) FLSR-A,B	(7) Fire Protection	
Loss of SFP Inventory[2][3]	S	(1)RHR-B[5]		(3)MUWC-A,B,C		(5)FLSS-A,B	(6) FLSR-A,B	(7) Fire Protection	(8) Isolation leakage
	A			(3)MUWC-A,C		(5)FLSS-A,B	(6) FLSR-A,B	(7) Fire Protection	(8) Isolation leakage
	C			(3)MUWC-B		(5)FLSS-A,B	(6) FLSR-A,B	(7) Fire Protection	(8) Isolation leakage
	D	(1)RHR-A		(3)MUWC-A,B,C		(5)FLSS-A	(6) FLSR-A,B	(7) Fire Protection	(8) Isolation leakage
	E			(3)MUWC-B		(5)FLSS-A,B	(6) FLSR-A,B	(7) Fire Protection	(8) Isolation leakage
	F			(3)MUWC-A,B,C	(4)SPCU	(5)FLSS-A,B	(6) FLSR-A,B	(7) Fire Protection	(8) Isolation leakage

[1]AC power source (EDG or BDG) are considered in front line systems except for FP and FLSR.
[2]MUWC automatic injection is additionally credited.
[3]When the large leak occurs, any makeup function cannot be credited.
[4]When the reactor building environment is degraded by some reactor fault such as ISLOCA, BOC and others, safety functions those are in R/B cannot be credited.
[5]RHRs are credited as makeup function (not credited as decay heat removal)

25.9.4 System Analysis

The System Analysis for the SFP PSA follows the same basic procedure as that of the Internal Events at Power PSA and the Shutdown PSA, see Section 25.4, 7 and 8.

System fault trees were derived by means of the following steps:

(1) Identifying the safety functions

(2) Identifying the front-line safety systems

(3) Identifying support systems of the above front-line systems

(4) Developing linkage of fault trees

First, the safety functions that should be performed to prevent fuel damage were identified. The following safety functions were considered in the Level 1 SFP PSA:

- Reactivity Control
- Decay Heat Removal
- SFP Structural Integrity
- SFP Isolation
- SFP Water Make-Up

Second, front-line safety systems that perform the required safety functions were identified. Front-line safety systems credited in sequence analysis task are shown in Table 25.9.4-1.

Third, support systems for front-line safety systems were identified. Operation of front-line systems requires the support of other systems such as electrical power, component cooling and room cooling.

Finally, fault trees of support systems were created for each and linked at the component level with each other.

Table 25.9.4-1 Systems to be Modelled in Internal Events SFP PSA

Function	System		System Code
Heat removal	RHR-FPC Assist mode		E11
	Fuel Pool Cooling and Clean-up System (FPC)		G41
Water injection	Residual Heat Removal system (RHR)		E11
	Make-up Water Condensate System (MUWC)		P13
	Suppression Pool Clean-up System (SPCU)		G51
	Fire Protection System(FP)		U43
	Flooding system of Specific Safety Facility (FLSS)		E71
	Flooding system of Reactor Building (FLSR)		E72
Support System	Electrical Power Distribution system (R10)	AC Power Supply	-
		- Metal-Clad Switchgear (M/C)	R22
		- Power Centre (P/C)	R23
		- Motor Control Centre (MCC)	R24
		Emergency Diesel Generator system (EDG)	R43
		Alternative Generator system (A/G)	R44
		DC Power Supply	R42
	Reactor Building Cooling Water System (RCW)		P21
	Reactor Building Service Water System (RSW)		P41
	Emergency Equipment Cooling Water System(EECW)		P27
	Heating Ventilating and Air Conditioning System (HVAC)		U41
	HVAC Normal Cooling Water System (HNCW)		P24
	HVAC Emergency Cooling Water System (HECW)		P25
	I & C (including digital system)		A32
	Instrument Air System (IA)		P52
	Station Service Air System (SA)		P51

25.9.5 Level 1 Quantification and Results

This section summarises the quantification of Internal Events SFP Level 1 PSA for the UK ABWR and documents the calculation of FDF due to internal events in the SFP for all POSs, including full power and shutdown.

25.9.5.1 Model Results Summary

The UK ABWR PSA model consists of event trees and fault trees that are quantified using a fault tree linking process. The calculation of the total SFP FDF is performed as a single top gate.

As a result of quantification, the total SFP FDF is 4.16E-07 /y. Details of the results are described in the next subsections.

25.9.5.2 Significant Contributors to SFP FDF

(1) Significant POSs

Figure 25.9.5-1 shows a summary contribution in the form of a pie chart. SFP FDFs for POSs are shown below.

- POS S : 1.18E-08 /y
- POS A : 2.43E-07 /y
- POS C : 3.05E-08 /y
- POS D : 9.38E-08 /y
- POS E : 5.11E-09 /y
- POS F : 3.17E-08 /y

POS A has the highest FDF (about 58 percent of the total SFP FDF). The initiating event “Loss of Class 1 AC due to human error” in POS A contributes to about 49 percent of the total SFP FDF. It should be noted that this type-B human error is modelled in POS A. Risk insights and possible measures are described in section 25.9.5.4.

The second contributor is POS D (about 23 percent of the total SFP FDF). The initiating event “LOCA-CUW” contributes to about 19 percent of the total SFP FDF. In POS D, one division of the backup building systems is out of service.

Despite the highest decay heat, POS E (full core offload) has the smallest FDF due to having the most redundant mitigation systems.

It should be noted that the SFP Level 1 PSA does not includes the POS B condition. This plant risk (POS B) is addressed by the shutdown PSA (Section 25.8). The POS B case of SFP fuel damage only is assigned to fuel damage sequence PDS-VIII, which is then used in the SFP Level 2 PSA [Ref-25.76].

(2) Significant Initiating Events

Table 25.9.5-1 shows the detailed contribution of initiating events to the SFP FDF.

The initiating events that have the largest contributors to SFP FDF are identified to provide a perspective on the results.

Loss of Class 1 AC due to type-B human error is the highest contributor. It has approximately 49 percent to the total SFP FDF. In addition, Loss of Class 1 AC due to non-HFE has about 6.1 percent contribution. This IE disables FPC-B, RHR-B and MUWC-B, such that only FLSS and FLSR are available for low pressure make-up.

LOCA-CUW in POS D is the second highest contributor: approximately 19 percent to the total SFP FDF. This reactor challenge IE disables all safety measures installed in R/B.

Generic LOOP IEs contribute to less than 1 percent of the total SFP FDF.

The heavy load drop IE contributes less than 0.1 percent to the SFP FDF.

Table 25.9.5-1 FDF Contribution by Initiating Event

Rank	Norm IE	Description	Contribution
1	___-HFE-SB-AC_A	Loss of a Class 1 AC bus	48.9 %
2	IE-LOCA-CUW-O_D	LOCA at CUW outside PCV at POS D	19.2 %
3	IE-LOCA-CUW-O_A	LOCA at CUW outside PCV at POS A	4.8 %
4	IE-AC-C1_A	Loss of AC Class 1E Bus Support System Initiating Event at POS A	3.2 %
5	IE-LOCA-CUW-O_C	LOCA at CUW outside PCV at POS C	3.2 %
6	IE-LOCA-RHR-SLI_D	LOCA at RHR suction line inside PCV at POS D	2.8 %
7	IE-AC-C1_C	Loss of AC Class 1E Bus Support System Initiating Event at POS C	2.5 %
8	IE-LOCA-CUW-O_S	LOCA at CUW outside PCV at POS S	2.4 %
9	IE-RHR-SFP	Loss of RHR SFP cooling mode	0.9 %
10	IE-LOCA-RHR-SLI_A	LOCA at RHR suction line inside PCV at POS A	0.7 %
11	IE-FPCB_F	Loss of FPC-B Support System Initiating Event at power	0.6 %
12	IE-LOCA-RHR-SLI_C	LOCA at RHR suction line inside PCV at PSO C	0.5 %
13	IE-HVAC-RBEEEAZ_A	Loss of R/BEEE/Z HVAC HVAC (A) Support System Initiating Event at POS A	0.4 %
14	IE-LOCA-RHR-SLI_S	LOCA at RHR suction line inside PCV at POS S	0.4 %
15	IE-HVAC-RBEEEBZ_C	Loss of R/BEEE/Z HVAC HVAC (B) Support System Initiating Event at POS C	0.3 %
16	IE-AC-C1_D	Loss of AC Class 1E Bus Support System Initiating Event at POS D	0.3 %
17	IE-FPCB_C	Loss of FPC-B Support System Initiating Event at POS C	0.3 %
18	IE-RCWB_SFP_C	Loss of RCW/RSW-B Support System Initiating Event for SFP PSA at POS C	0.3 %
19	IE-HNCW_D	Loss of HNCW Initiating Event at POS D	0.1 %
20	TEG_POS-E	LOOP Grid Related at POS E	0.1 %
21	IE-HNCW_C	Loss of HNCW Initiating Event at POS C	0.1 %
22	TES_POS-E	LOOP Switchyard Centred at POS E	0.1 %
23	IE-HNCW_A	Loss of HNCW Initiating Event at POS A	0.1 %
24	TEW_POS-E	LOOP Weather Related at POS E	0.1 %
25	IE-FPCA_A	Loss of FPC-A Support System Initiating Event at POS A	0.1 %
26	IE-AC-C1_S	Loss of AC Class 1E Bus Support System Initiating Event at POS S	0.1 %
27	IE-AC-C3_A	Loss of single Class 3 AC Bus Support System Initiating Event at POS A	0.1 %
28	IE-AC-C3_C	Loss of single Class 3 AC Bus Support System Initiating Event at POS C	0.1 %
29	IE-RCWA_SFP_A	Loss of RCW/RSW-A Support System Initiating Event for SFP PSA at POS A	0.1%

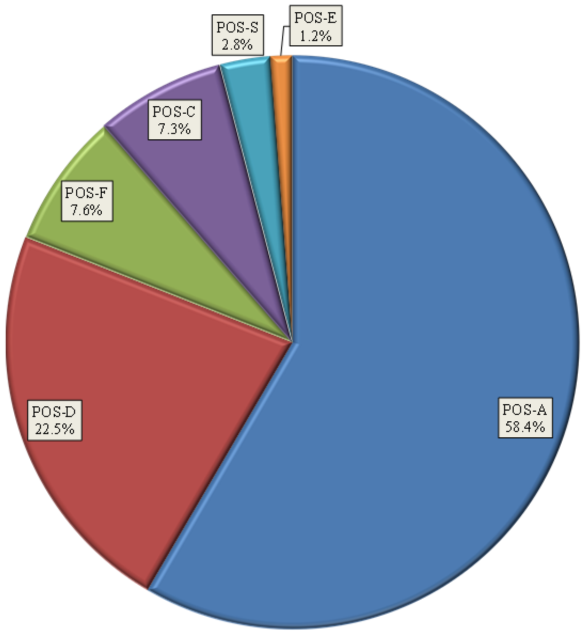

Figure 25.9.5-1 Contribution to SFP FDF by POS

(3) Significant Accident Sequences

Table 25.9.5-2 summarises all quantified accident sequences, the corresponding frequencies, and the percentage contribution to SFP FDF.

Figure 25.9.5-2 illustrates the SFP FDF contribution for the top ten accident sequences in the form of a pie chart. Those accident sequences which contribute more than 1 percent to the SFP FDF are described below.

Rank 1: <u>SFPC-A04</u>

SFPC-A04 is a loss of SFP cooling sequence in POS A.

- The Operating FPC-A fails due to various causes due to the IE.
- Operator diagnosis succeeds to detect the loss of SFP cooling.
- SFP makeup by MUWC-A through RHR-B and FPC-B lines or by Fire Protection system injection fails.
- Alternative make-up from FLSS or FLSR fails.
- With no make-up, fuel uncovery eventually occurs resulting in fuel damage after about 303 hours after IE occurring.

This accident sequence contributes 53 percent of total SFP FDF.

Rank 2: SFPC-D_LOCA3

SFPC-D_LOCA is a loss of SFP cooling sequence in POS D caused by effects of LOCA in the reactor outside of containment (ISLOCA or BOC as reactor challenge).

- The Operating FPC-B fails due to the effects of the LOCA.
- This is a reactor challenge scenario. No safety system is credited whose components are located in the R/B, or whose alignment requires human actions in the R/B. This effectively fails: FPC-A (B), MUWC-A (B, C), RHR SFP mode and Fire Protection system.
- Operator diagnosis succeeds to detect the loss of SFP cooling
- With the normal make-up systems failed or unavailable due to the effects of the LOCA, only alternative make-up from FLSS or FLSR is available, but these also fail.
- With no make-up, fuel uncovery eventually occurs resulting in fuel damage after about 303 hours after IE occurring.

This accident sequence contributes 22 percent of total SFP FDF.

Rank 3: SFPC-A_LOCA3

SFPC-A_LOCA is a loss of SFP cooling sequence in POS A caused by effects of LOCA in the reactor outside of containment (ISLOCA or BOC as reactor challenge).

- The operating FPC-A fails due to the effects of the LOCA.
- This is a reactor challenge scenario. No safety system is credited whose components are located in the R/B, or whose alignment requires human actions in the R/B. This effectively fails: FPC-A (B), MUWC-A (B, C), RHR SFP mode and Fire Protection system.
- Operator diagnosis succeeds to detect the loss of SFP cooling
- With the normal make-up systems failed or unavailable due to the effects of the LOCA, only alternative make-up from FLSS or FLSR is available, but these also fail.
- With no make-up, fuel uncovery eventually occurs resulting in fuel damage after about 303 hours later than IE occurring.

This accident sequence contributes 5.5 percent of total SFP FDF.

Rank 4: SFPC-F_LOCA3

SFPC-F_LOCA is a loss of SFP cooling sequence in POS F caused by effects of LOCA in the reactor outside of containment (ISLOCA or BOC as reactor challenge).

- The operating FPC-B fails due to the effects of the LOCA.
- This is a reactor challenge scenario. No safety system is credited whose components are located in the R/B, or whose alignment requires human actions in the R/B. This effectively fails: FPC-A (B), MUWC-A (B, C), RHR SFP mode and Fire Protection system.
- Operator diagnosis succeeds to detect the loss of SFP cooling

- With the normal make-up systems failed or unavailable due to the effects of the LOCA, only alternative make-up from FLSS or FLSR is available, but these also fail.
- With no make-up, fuel uncovery eventually occurs resulting in fuel damage after about 303 hours later than IE occurring.

This accident sequence contributes 4.1 percent of total SFP FDF.

Rank 5: SFPC-C_LOCA3

SFPC-C_LOCA is a loss of SFP cooling sequence in POS C caused by effects of LOCA in the reactor outside of containment (ISLOCA or BOC as reactor challenge).

- Event sequence is similar to SFPC-D_LOCA03 except for the credit of FLSS/FLSR. Although only one division of backup building equipment can be credited in POS D, redundancy of these systems can be credited in POS C.

This accident sequence contributes 3.8 percent of total SFP FDF.

Rank 6: SFPC-C04

SFPC-C04 is a loss of SFP cooling in POS C.

- The operating FPC-B fails due to various causes due to the IE.
- Operator diagnosis succeeds to detect the loss of SFP cooling.
- SFP make-up by MUWC-B through RHR-B and FPC-B lines or by Fire Protection system injection fails.
- Alternative make-up from FLSS or FLSR fails.
- With no make-up, fuel uncovery eventually occurs resulting in fuel damage after about 303 hours later than IE occurring.

This accident sequence contributes 3.4 percent of total SFP FDF.

Rank 7: SFPC-S_LOCA3

SFPC-S_LOCA is a loss of SFP cooling sequence in POS S caused by effects of LOCA in the reactor outside of containment (ISLOCA or BOC as reactor challenge).

- Event sequence is similar to SFPC-D_LOCA03. The difference is that the credited safety division is opposite and redundant division of backup building systems can be credited in POS S.

This accident contributes 2.8 percent of total SFP FDF.

Rank 8: SFPC-F05

SFPC-F05 is a boil-off sequence in POS F.

- The operating FPC-B fails due to various causes due to the IE.

- Operator diagnosis succeeds to detect the loss of SFP cooling, but the standby FPC-A train fails.
- SFP make-up by MUWC through RHR and FPC lines, by FP or by SPCU fails.
- Alternative make-up from FLSS or FLSR fails.
- With no make-up, fuel uncovery eventually occurs resulting in fuel damage after about 303 hours later than IE occurring.

This accident sequence contributes 2.1 percent of total SFP FDF.

Rank 9: SFPLOOP-F05

SFPLOOP-F05 is a LOOP sequence in POS F.

- LOOP occurs causing a loss of the running FPC pump.
- Standby EDG succeeds to restore AC power to cooling systems.
- Restart of any FPC train fails, resulting in heatup of SFP and boiling.
- Operator diagnosis succeeds to detect low water level in SFP.
- SFP makeup by MUWC through the RHR and FPC lines, by Fire Protection system injection, or injection by SPCU fails.
- Alternative make-up from FLSS or FLSR fails
- With no make-up, fuel uncovery eventually occurs resulting in fuel damage after about 303 hours after IE occurring.

This accident sequence contributes 1.0 percent of total SFP FDF.

Rank 10: SFPC-E04

SFPC-E04 is a loss of SFP cooling sequence with fuel damage in POS E (full core offload).

- Operating RHR-B on SFP cooling mode fails due to various causes due to the IE.
- Operator diagnosis succeeds to detect the loss of RHR cooling for the SFP.
- SFP make-up by MUWC-B via RHR-B and FPC-B lines or by FP fails.
- Alternative make-up from FLSS or FLSR fails.
- With no make-up, fuel uncovery eventually occurs resulting in fuel damage after about 104 hours later than IE occurring.

This accident sequence contributes 0.9 percent of total SFP FDF.

Table 25.9.5-2 All SFP Fuel Damage Sequences

Rank	Event	POS	Probability	Contribution to SFP POS FDF
1	SFPC-A04	POS A	2.19E-07	52.7 %
2	SFPC-D_LOCA03	POS D	9.17E-08	22.0 %
3	SFPC-A_LOCA03	POS A	2.30E-08	5.5 %
4	SFPC-F_LOCA03	POS F	1.69E-08	4.1 %
5	SFPC-C_LOCA03	POS C	1.56E-08	3.8 %
6	SFPC-C04	POS C	1.40E-08	3.4 %
7	SFPC-S_LOCA03	POS S	1.15E-08	2.8 %
8	SFPC-F05	POS F	8.92E-09	2.1 %
9	SFPLOOP-F05	POS F	4.30E-09	1.0 %
10	SFPC-E04	POS E	3.91E-09	0.9 %
11	SFPC-D06	POS D	1.84E-09	0.4 %
12	SFPIL-F08	POS F	1.46E-09	0.4 %
13	SFPLOOP-A05	POS A	7.49E-10	0.2 %
14	SFPLOOP-E05	POS E	7.20E-10	0.2 %
15	SFPLOOP-C11	POS C	5.66E-10	0.1 %
16	SFPC-S06	POS S	3.04E-10	0.1 %
17	SFPLOOP-D06	POS D	2.96E-10	0.1 %
18	SFPLOOP-E20	POS E	2.41E-10	0.1 %
19	SFPLOOP-C05	POS C	2.37E-10	0.1 %

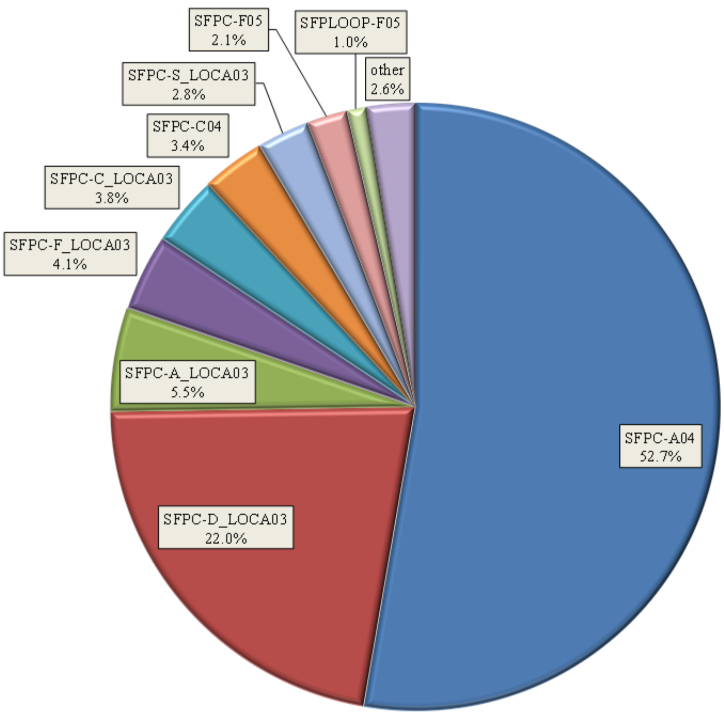

Figure 25.9.5-2 Contribution to SFP FDF by Accident Sequences

(4) Significant Accident Classes

All SFP fuel damage is assigned to Accident Class VII, fuel damage sequence PDS-VIII.

25.9.5.3 Importance Analysis

The following discussions detail the safety related importance of various systems, components and operator actions.

(1) Component important to safety (High Fussell-Vesely Importance)

Table 25.9.5-4 shows the importance of components. The components important to safety are discussed here ranked according to F-V (a measure of how much a component's reliability directly influences the FDF). The components are listed in order of risk significance. A high F-V importance indicates that risk can be reduced by improving the component's reliability and/or availability.

<u>FLSS initiation</u>
The highest F-V event is ___-HFE-FC-FL "Failure of manual initiation of FLSS": The F-V is 9.62E-01. This is due to the FLSS importance in mitigating every accident sequence. FLSS can cope with all accident sequences except for SFP Large leak due to SFP catastrophic failure.

<u>FLSR initiation</u>
The second highest F-V event is FLSR-SD "FLSR (Mobile Injection Facility) Unavailability": The F-V is 9.01E-01. FLSR provides a similar function as FLSS, and results in a similar F-V importance. FLSR can cope with all accident sequences except for SFP Large leak due to SFP catastrophic failure.

<u>Class 1 AC bus</u>
The third highest F-V event is ___-HFE-SB-AC_A "Loss of a Class 1 AC bus" due to incorrect operation: the F-V is 4.89E-01. This event leads to unrecoverable loss of AC power.

(2) Significant CCF events (FV > 0.005)

CCF events listed in Table 25.9.5-5 are significantly lower in frequency compared to similar cutsets with the total CCF of all components. If the CCF of the partial group is set to 1.0 (for RAW), then these cutsets have a significant increase in their frequency, which drives the high worth for RAW.

(3) Significant Post-Initiator HFEs (FV > 0.005 or RAW > 2.0)

Manual initiation of FLSS, FLSR and Fire Protection system listed in Table 25.9.5-6 have high F-V importance because redundancy of Class 1 system is degraded. Except FLSS, FLSR and loss of Class 1 AC, the FV importance of post-initiator HFEs still remain about 0.3 (___-HFE-MV-FP) and about 0.2 (___-HFE-FC-FP) for the initiation of Fire Protection system.

FLSR failure probability is represented by the associated human error probabilities (FLSR-SD, FLSR-SD_ST) which are based on shutdown PSA condition as described in the following paragraph. It is assumed FLSR is required to be in service (connected to the injection point) before starting POS B-2 and until completion of POS C [Ref-25.75]. This enables credit of FLSR in the accident sequences which do not have the time window more than 8 hours for FLSR In other POSs, FLSR is credited for the accident sequences which have the time window for at least 8 hours for FLSR [Ref-25.75]. These HFEs have high importance values because they are independent from Class 1 support systems.

(4) Significant Pre-Initiator HFEs (FV > 0.005 or RAW > 2.0)

F-V importance of the top two events listed in Table 25.9.5-7 are lower than 0.005. These events disable FLSS (both two trains) however which have small F-V importance.

Manual valve left close E71-F032 or F035 are related to FLSS injection.

Manual valve P13-MVF099 Left Close has RAW rank 3. This human error disables Fire Protection system injection via MUWC due to disconnection of MUWC and Fire Protection system.

(5) Initiating Event HFEs (TYPE B)

The human error "___-HFE-SB-AC_A" causing initiating events have high FV (see Table 25.9.5-8). The reason is that this initiator has wider impacts on the mitigation systems designed as Class 1.

(6) System Level Importance

System importance was manually calculated All the basic events that have the unique system ID (including pre-initiator HFEs) are assigned to one of the "systems" specified by the system ID. Some miscellaneous basic events, e.g., BBG-1, BBG-2, and FLSR unavailability are manually checked for specifying the relevant systems. The CCF events and post initiator HFEs are not included. The initiating event basic events are also not included.

Notes:

- Some minor systems, e.g., D11 "safety process radiation monitoring system" in which only a sensor is modelled as the source of spurious signal, is not shown.
- In this system level importance discussion, "AC" means Atmospheric control system. It has a small FV because the system has extremely small risk contribution.
- AC power is delivered through Metal-Clad switchgear (M/C), Power centre (P/C) and Motor Control Centre (MCC). Although these are not name of systems, these are in the table because it is useful to understand the importance analysis.
- The RHR is modelled both in FPC assist mode operation, and also as an initiating event (LOCA outside containment).

System level importance measures are summarised in Table 25.9.5-3. The following Class 3 systems have relatively high FV or RAW importance:

- CUW: this is related to "LOCA outside containment" (reactor challenge). Therefore, FV and RAW are significant.
- Fire Protection System (FP)/MUWC: These systems have injection function in many cases. Therefore, they have a high F-V.
- HNCW/TCW: HNCW/TCW are related to cooling of electrical equipment and LOOP initiating event.
- IA is related to FPC continuous operation and it related to loss of SFP cooling initiating event.
- SPCU/CRD: Although SPCU can be credited as water injection function, the effective POS is limited. The reason for a high RAW is caused by the loss of inventory initiating event for

shutdown reactor PSA (reactor challenge). CRD is also related to the same initiating event (reactor challenge).

- Both FLSS and FLSR have high F-V importance. FLSS depends on the BBGs and its initiation signals. On the other hand, FLSR has no dependency between those supporting system and C&I. Therefore, F-V importance of FLSR is higher than that of FLSS.

Table 25.9.5-3 System Level Importance

Rank (F-V)	SYSTEMS	F-V	RAW
1	FLSR	9.84E-01	4.31E+01
2	CUW	2.92E-01	2.95E+04
3	RHR	6.91E-02	1.20E+04
4	CONTROL_PANELS	6.42E-02	4.33E+04
5	FP	5.92E-02	1.12E+01
6	PC	5.51E-02	1.09E+05
7	MC	2.86E-02	3.90E+04
8	FLSS	2.62E-02	1.03E+03
9	MUWC	2.18E-02	1.56E+01
10	HVAC	1.10E-02	7.60E+02
11	FPC	1.08E-02	3.99E+04
12	NB	1.03E-02	3.40E+04
13	DG	9.11E-03	4.31E+00
14	HPCF	7.06E-03	4.76E+03
15	RCW	6.53E-03	1.12E+02

Table 25.9.5-4 Importance of Components (sorted by F-V) (1/2)

Rank	Event	F-V	RAW	Description
1	___-HFE-FC-FL	9.62E-01	2.41E+01	Failure of manual initiation of FLSS
2	FLSR-SD	9.01E-01	3.47E+01	FLSR (Mobile Injection Facility) Unavailability
3	@POS-A	5.85E-01	1.00E+00	During POS A
4	SQSF_SFPC-A04	5.27E-01	1.00E+00	sequence tag for SFPC-A04
5	___-HFE-SB-AC_A	4.90E-01	1.44E+02	Loss of a Class 1 AC bus
6	PLANT-UNAVAILABILITY	4.22E-01	6.60E+00	Plant Unavailability Factor - Shutdown
7	___-HFE-MV-FP	2.95E-01	1.16E+01	Failure of opening FP valve
8	@POS-D	2.26E-01	1.00E+00	During POS D
9	WEIGHT-FACTOR-D	2.25E-01	1.60E+00	Weight Factor at POS D
10	SQSF_SFPC-D_LOCA03	2.20E-01	1.00E+00	sequence tag for SFPC-D_LOCA03
11	IE-LOCA-CUW-O_D	1.92E-01	8.09E-01	LOCA at CUW outside PCV at POS D
12	___-HFE-FC-FP	1.85E-01	1.17E+01	Failure of manual initiation of FP
13	WEIGHT-FACTOR-A	9.48E-02	2.21E+00	Weight Factor at POS A
14	FLSR-SD_ST	8.30E-02	9.39E+00	FLSR (Mobile Injection Facility) Unavailability (Line-up before IE)
15	@POS-F	7.60E-02	1.00E+00	During POS F
16	@POS-C	7.33E-02	1.00E+00	During POS C
17	WEIGHT-FACTOR-C	7.30E-02	1.43E+00	Weight Factor at POS C
18	PLANT-AVAILABILITY	7.07E-02	1.01E+00	Plant Availability Factor - At power
19	SQSF_SFPC-A_LOCA03	5.53E-02	1.00E+00	sequence tag for SFPC-A_LOCA03
20	___-HFE-CB-SE	5.37E-02	1.17E+01	Circuit breaker selection error
21	IE-LOCA-CUW-O_A	4.83E-02	9.52E-01	LOCA at CUW outside PCV at POS A
22	SQSF_SFPC-F_LOCA03	4.05E-02	1.00E+00	sequence tag for SFPC-F_LOCA03
23	FACTOR_RBS_OK	4.02E-02	1.00E+00	Reactor challenge (R/B Steaming)
24	FL_PDS-OK2	3.92E-02	1.00E+00	Sequence tag for L1--OK2
25	SQSF_SFPC-C_LOCA03	3.76E-02	1.00E+00	sequence tag for SFPC-C_LOCA03
26	SQSF_SFPC-C04	3.37E-02	1.00E+00	sequence tag for SFPC-C04
27	U43-EDP-RA-__-C003	3.32E-02	1.60E+00	Diesel Engine-Driven Pump C003 Fail to Run > 1 Hour
27	U43-EDP-RA-__-C004	3.32E-02	1.60E+00	Diesel Engine-Driven Pump C004 Fail to Run > 1 Hour
29	IE-AC-C1_A	3.25E-02	9.68E-01	Loss of AC Class 1E Bus Support System Initiating Event at POS A
30	IE-LOCA-CUW-O_C	3.20E-02	9.68E-01	LOCA at CUW outside PCV at POS C
31	@POS-S	2.84E-02	1.00E+00	During POS S

Table 25.9.5-4 Importance of Components (sorted by F-V) (2/2)

Rank	Event	F-V	RAW	Description
31	WEIGHT-FACTOR-S	2.84E-02	1.75E+00	Weight Factor at POS S
33	IE-LOCA-RHR-SLI_D	2.79E-02	9.72E-01	LOCA at RHR suction line inside PCV at POS D
34	SQSF_SFPC-S_LOCA03	2.76E-02	1.00E+00	sequence tag for SFPC-S_LOCA03
35	IE-AC-C1_C	2.51E-02	9.75E-01	Loss of AC Class 1E Bus Support System Initiating Event at POS C
36	IE-LOCA-CUW-O_S	2.41E-02	9.76E-01	LOCA at CUW outside PCV at POS S
37	SQSF_SFPC-F05	2.12E-02	1.00E+00	sequence tag for SFPC-F05
38	U43-EDP-RA-FP_1_2	2.00E-02	1.18E+01	CCF of two components: U43-EDP-RA-__-C003 & U43-EDP-RA-__-C004
39	G31-MPR-EL-__-C001B	1.70E-02	2.95E+04	Motor-Driven Pump (Running) C001B External Leak Large
40	G31-HX_-SL-__-B001A	1.66E-02	2.95E+04	Heat Exchanger B001A Shell External Leak Large
40	G31-HX_-SL-__-B001B	1.66E-02	2.95E+04	Heat Exchanger B001B Shell External Leak Large
40	G31-HX_-SL-__-B001C	1.66E-02	2.95E+04	Heat Exchanger B001C Shell External Leak Large
43	G31-MPR-EL-__-C001A	1.49E-02	2.59E+04	Motor-Driven Pump (Running) C001A External Leak Large
44	E11-MV_-EL-__-F009A	1.30E-02	2.95E+04	Manual Valve F009A External Leak Large
44	E11-MV_-EL-__-F009B	1.30E-02	2.95E+04	Manual Valve F009B External Leak Large
44	E11-MV_-EL-__-F009C	1.30E-02	2.95E+04	Manual Valve F009C External Leak Large
44	G31-MV_-EL-__-F004A	1.30E-02	2.95E+04	Manual Valve F004A External Leak Large
44	G31-MV_-EL-__-F004B	1.30E-02	2.95E+04	Manual Valve F004B External Leak Large
44	G31-MV_-EL-__-F005A	1.30E-02	2.95E+04	Manual Valve F005A External Leak Large
44	G31-MV_-EL-__-F005B	1.30E-02	2.95E+04	Manual Valve F005B External Leak Large
44	G31-MV_-EL-__-F006A	1.30E-02	2.95E+04	Manual Valve F006A External Leak Large
44	G31-MV_-EL-__-F006B	1.30E-02	2.95E+04	Manual Valve F006B External Leak Large
44	G31-MV_-EL-__-F007B	1.30E-02	2.95E+04	Manual Valve F007B External Leak Large
44	G31-MV_-EL-__-F009A	1.30E-02	2.95E+04	Manual Valve F009A External Leak Large
44	G31-MV_-EL-__-F009B	1.30E-02	2.95E+04	Manual Valve F009B External Leak Large
44	G31-MV_-EL-__-F011	1.30E-02	2.95E+04	Manual Valve F011 External Leak Large
44	G31-MV_-EL-__-F012	1.30E-02	2.95E+04	Manual Valve F012 External Leak Large
44	G31-MV_-EL-__-F013	1.30E-02	2.95E+04	Manual Valve F013 External Leak Large
59	H11-IF_-UF-__-L1C2	1.28E-02	1.96E+00	Interface L1C2 Undetectable Loss of Function
60	E71-SR_-PG-__-D009	1.26E-02	1.03E+03	Strainer D009 Plugged

Table 25.9.5-5 Importance of CCF (sorted by F-V)

CCF Event	F-V	Description
U43-EDP-RA-FP_1_2	2.00E-02	CCF of two components: U43-EDP-RA-__-C003 & U43-EDP-RA-__-C004
___-SLU-LF-__-PCIS	2.41E-03	Software Failure induced CCF PCIS Loss of Function
___-SLU-LF-__-HVAC	2.03E-03	Software Failure induced CCF HVAC Loss of Function
U41-FNR-R_-C804_1_2	1.91E-03	CCF of two components: U41-FNR-R_-__-C804A & U41-FNR-R_-__-C804B
U41-FNR-R_-C805_1_2	1.91E-03	CCF of two components: U41-FNR-R_-__-C805A & U41-FNR-R_-__-C805B
___-SLU-LF-__-RHR_RCW	1.55E-03	Software Failure induced CCF RHR_RCW Loss of Function
___-SLU-LF-__-RPS	1.40E-03	Software Failure induced CCF RPS Loss of Function
U43-EDP-FS-FP_1_2	1.17E-03	CCF of two components: U43-EDP-FS-__-C003 & U43-EDP-FS-__-C004

Table 25.9.5-6 Importance of Post –Initiator HFEs (1/2 sorted by F-V)

HFE	F-V	Description
___-HFE-FC-FL	9.62E-01	Failure of manual initiation of FLSS
FLSR-SD	9.01E-01	FLSR (Mobile Injection Facility) Unavailability
___-HFE-MV-FP	2.95E-01	Failure of opening FP valve
___-HFE-FC-FP	1.85E-01	Failure of manual initiation of FP
FLSR-SD_ST	8.31E-02	FLSR (Mobile Injection Facility) Unavailability (Line-up before IE)
___-HFE-CB-SE	5.37E-02	Circuit breaker selection error
___-HFE-FC-SP	5.84E-03	Failure of manual initiation of SPCU
___-HFE-FC-FC	2.09E-03	Failure of manual initiation of FPC
___-HFE-FC-MU	1.24E-03	Failure of manual initiation of MUWC

Table 25.9.5-6 Importance of Post –Initiator HFEs (2/2 sorted by RAW)

HFE	RAW	Description
FLSR-SD	3.47E+01	FLSR (Mobile Injection Facility) Unavailability
___-HFE-FC-FL	2.41E+01	Failure of manual initiation of FLSS
___-HFE-FC-FP	1.17E+01	Failure of manual initiation of FP
___-HFE-CB-SE	1.17E+01	Circuit breaker selection error
___-HFE-MV-FP	1.16E+01	Failure of opening FP valve
FLSR-SD_ST	9.39E+00	FLSR (Mobile Injection Facility) Unavailability (Line-up before IE)

Table 25.9.5-7 Importance of Pre –Initiator HFEs (1/2 sorted by F-V)

HFE	FV	Description
E71-HFE-CL-__-MVF032	2.09E-03	Manual valve F032 left close
E71-HFE-CL-__-MVF035	2.09E-03	Manual valve F035 left close

Table 25.9.5-7 Importance of Pre –Initiator HFEs (2/2 sorted by RAW)

HFE	RAW	Description
E71-HFE-CL-__-MVF032	2.42E+01	Manual valve F032 left close
E71-HFE-CL-__-MVF035	2.42E+01	Manual valve F035 left close
P13-HFE-CL-__-MVF099	1.15E+01	Manual Valve F099 Left Closed
E71-HFE-CL-__-MVF005A	6.19E+00	Manual valve F005A left close
E71-HFE-CL-__-MVF001A	6.16E+00	Manual valve F001A left close
E71-HFE-CL-22-MVF002AC	6.12E+00	CCF of Manual valve F002AC left close
E71-HFE-CL-22-MVF004AC	6.12E+00	CCF of Manual valve F004AC left close
E71-HFE-CL-22-MVF071AB	6.12E+00	CCF of Manual valve F071AB left closed
B21-HFE-MC-44-PT009ABCD	3.07E+00	CCF Miscalibration of Pressure Transmitter B21-PT009ABCD

Table 25.9.5-8 Initiating Event HFEs (TYPE B)

HFE	F-V	Description
___-HFE-SB-AC_A	4.89E-01	Loss of a Class 1 AC bus
___-HFE-FB-FC	1.00E-05	Loss of SFP cooling (Configuration control)

25.9.5.4 PSA Results and Insights

The quantification provides the following results:

- POS A has relatively high risk (about 58 percent of the total SFP FDF) because Class 1 division 2 is out of service. Division 1 and 3 are available but division 3 effectiveness is limited for SFP critical safety function.
- POS D has relatively high risk (about 23 percent of the total SFP FDF) because one division of the backup building systems is out of service.
- Due to the partially degraded redundancy of the Class 1 systems, FLSS is important in all POS. The post-initiator HFE that has the highest F-V is the manual initiation of FLSS.
- FLSR provides considerable risk reduction. It is assumed FLSR is required to be connected to the injection point before an IE during POS C per the recommendation of experienced system engineers and maintenance planners in the Design Review meeting on December 27 2015, so that the redundancy of mitigation systems is improved under the condition Class 1 divisions 1 and 3 systems are in maintenance. This is consistent with shutdown PSA modelling.
- HFEs (Type B) on Loss of Class 1 AC initiating event has significant risk contribution in POS A. Review of this human failure event and refinement of this analysis may be effective to develop more realistic model when the information of this action is available.

The following insights were gained from the SFP PSA's dominant contributors to core and fuel damage risk:

<u>Systems which are important to have continued availability</u>

Based on the results of F-V importance analysis, the following systems are important to have continued availability.

- FLSR
- RHR
- Fire Protection system

Although CUW has also high F-V importance, this basic event is related to only ISLOCA/BOC initiating event. Therefore, it is not important to have continued availability.

<u>Systems whose degradation would lead to increased risk</u>

FLSR is essential to decrease the risk of HFEs (Type B) on Loss of Class 1 AC initiating event e.g. FLSR is aligned before the action for AC power switching.

<u>Design or procedural changes that could be enhanced based on further analysis</u>

Injection from Fire Protection system is one of the important countermeasures when LOOP occurs with loss of Class 1 AC power because Fire Protection system has a diesel driven pump that can be used to inject water without any AC power. However, the Fire Protection system is connected to MUWC using a motor operated valve thus it cannot be opened when the AC power is lost.

In the shutdown condition, only one safety division can be credited, as the other safety division is out of service due to maintenance. Recovery of the out of service safety division should be considered although this would likely involve a relatively long time margin.

25.9.6 Analysis for SFP Level 2 PSA

25.9.6.1 Definition of Large Release and Large Early Release

ASME PRA standard [Ref-25.13] gives the following definition.

Large early release: the rapid, unmitigated release of airborne fission products from the containment to the environment occurring before the effective implementation of off-site emergency response and protective actions such that there is a potential for early health effects.

Large early release frequency (LERF): expected number of large early releases per unit of time.

Based on the definition of LER, Large Release is defined as the unmitigated release of airborne fission products from the containment to the environment such that there is a potential significant health effects. It is judged adequate to treat all the fuel damage sequences except for spray success in Level 2 PSA as large release in the SFP Level 2 PSA.

The adequacy of this approach has been confirmed by comparing the calculated source terms with the definition of large release [Ref-25.86];

Once the CsI source term exceeds 10 %, the source term is large enough such that doses above early fatality threshold can sometimes occur within a population centre a few miles from the site.

The calculated release fraction of CsI exceeds 10 percent for all cases except for spray case (see Section 25.9.6.6 source term). Thus, it is judged adequate to include these release categories into the large release.

In conclusion, the Large Release Frequency (LRF) is the calculated as the sum of all the release category frequencies except for the sequence with success of SFP spray in Level 2 PSA.

CsI release fraction with the success of fuel cooling by spray is below 10 percent. Therefore, the release category of success of fuel cooling is excluded from the large release.

Among the fuel damage sequences with large release, those with large “early” release (LER) are defined considering the potential early health effects. The qualitative definition of LER is provided in the ASME PRA standard [Ref-25.13].

A more specific definition of LER is found in the EPRI PSA Applications Guide [Ref-25.87].

Unscrubbed Containment Failure Pathway of Sufficient Size to Release the Contents of the Containment (i.e., one volume change) Within One Hour, Which Occurs Before or Within 4 Hours of Vessel Breach

This definition cannot to be directly used in the SFP Level 2 PSA because the SFP is outside of the containment boundary. Alternatively, a large early release is defined as a large release which is expected to occur within 4 hours from an initiator.
LERF is also evaluated, considering the potential for early health effects. LERF category is defined as below.

- Sum of all frequencies of fuel heatup and damage sequences due to SFP drain down (large leak)

Late release categories are “fuel heatup and damage sequences due to drain down (small leak)” and “fuel heatup and damage sequences due to boil-off“. The timing of all calculated fission product releases for these release categories exceed 70 hours. Therefore, release categories of these release categories are excluded from the early release.

25.9.6.2 Level 1 PSA to Level 2 PSA interface

The SFP Level 2 PSA considers all fuel damage sequences from the SFP Level 1 PSA.

The scope of the SFP Level 2 PSA covers all scenarios following fuel damage in the SFP accident sequences which are quantified by the SFP Level 1 PSA and shutdown Level 1 PSA.

(1) Internal events SFP Level 1 PSA fuel damage sequence (POS S, A, C, D, E, F) including fuel damage sequence due to reactor challenge

(2) Internal events shutdown Level 1 PSA fuel damage sequence (POS B)

In the Level 2 SFP PSA, the above (1) i.e. fuel damage sequences result from three initiating event groups in the Level 1 SFP PSA (LOOP, Loss of SFP Cooling and Loss of SFP inventory) are defined as PDS and linked to Level 2 event tree and then quantified. In addition, the above (2) i.e. fuel damage sequences result from Level 1 shutdown PSA POS B defined as PDS VIII (fuel damage in only SFP) are also linked to Level 2 event tree. POS-B accident class VI in the shutdown PSA also includes fuel damage in SFP. In this accident class, spray is not credited and then probabilistically addressed in shutdown Level 2 PSA. The source term analysis is performed in SFP Level 2 PSA and the information is shared with shutdown Level 2 PSA.

In the SFP PSA, "Reactor challenge" is considered. The treatment for the "Reactor challenge" is explained as follows. Figure 25.9.6-1 shows the calculation stream of this SFP Level 2 PSA.

Both "Reactor challenge 1" and "Reactor challenge 2" are the links between reactor sequences and SFP sequences. The fuel damage sequences are linked to Level 2 SFP PSA. In addition, each POS in the shutdown condition also considers Reactor challenge in Level 1 SFP PSA. The resulting fuel damage sequence are linked to Level 2 SFP PSA

One Accident Class "Fuel Damage" is defined in this analysis. The accident class is taken over as PDSs in Level 2 PSA.

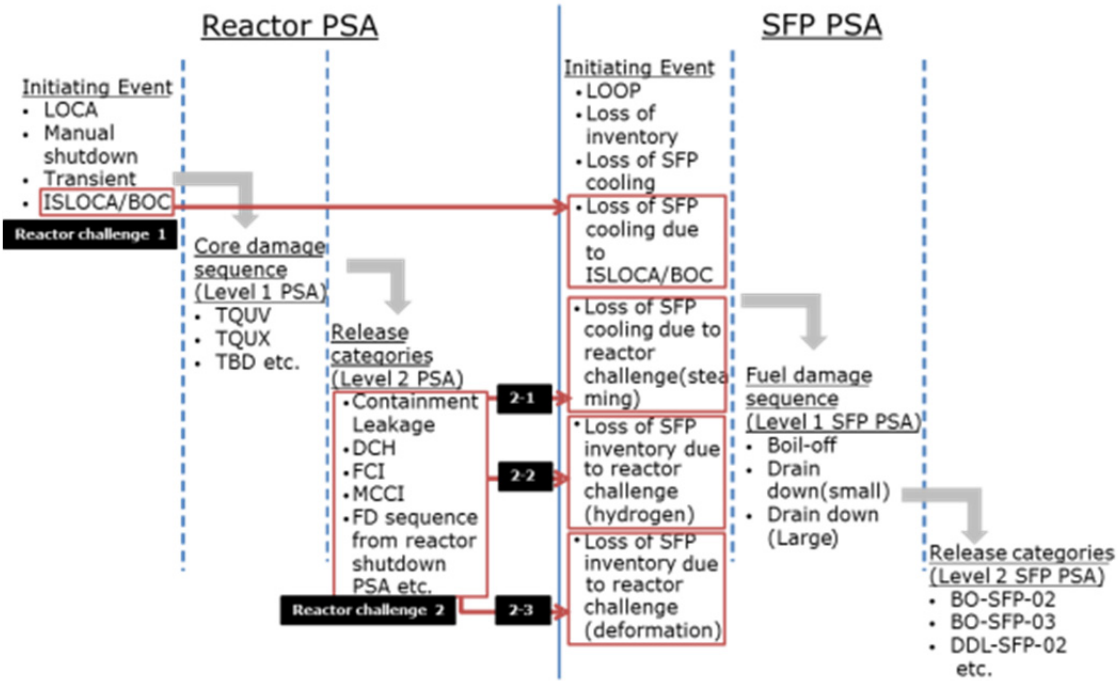

Figure 25.9.6-1 Calculation Stream in the Quantification Binning of Level 1 Plant Damage States

25.9.6.3 Other Model Changes from Level 1 PSA to Level 2 PSA

The event trees for Level 1 SFP PSA are not changed for the Level 2 PSA. FLSS operation is credited in Level 2 PSA with Level 2 PSA specific HFE. For the additional cues for operator actions in Level 2 PSA, instrumentation for a SFP severe accident is modelled.

25.9.6.4 Deterministic Accident Progression Analysis

Severe Accident Progression and Phenomena

Accident Progression and Key Phenomena

Potential mechanisms for fission product release from the SFP due to a severe accident condition were identified to conduct the SFP Level 2 PSA assessment. For the UK ABWR, containment failure modes were already identified in IEAP Level 2 PSA and containment performance analysis, but the containment boundary does not prevent fission product releases since the SFP is outside of containment. In this section, each fission product release path due to accident progression is described.

Accident progression for an SFP severe accident is outlined in Figure 25.9.6-2 as the extended condition of Level 1 PSA fuel damage end states. Boxes with a red outline show release path with accident progression or severe accident phenomena.

In the fuel damage sequence of the SFP Level 1 PSA, spent fuel is heated up by its decay heat. This fuel heatup takes a long time compared to reactor accident sequences because of lower decay heat level than that immediately after the reactor SCRAM and the effect of heat exchange between fuel rods and surrounding steam. If sufficient water is sprayed successfully within the time margin before the fuel heatup, a large fission product release can be prevented.

If spray operation fails to provide sufficient cooling, overheated fuel generates hydrogen by the reaction with surrounding steam and it accumulates on the operating deck. The blow out panel can be opened by manual operator action or a passive mechanism for hydrogen management. If the blow out panel is opened successfully, hydrogen can be vented but fission products are also released to the environment.

In most cases, hydrogen combustion on the operating deck is prevented by steam inerting, with steam generated from the SFP boiling. In some cases, the blowout panel is opened by active or passive mechanism. If hydrogen combustion occurs in spite of success of hydrogen management measures, the boundary of the secondary containment is damaged and fission products are released to the environment.

In more severe cases, the hydrogen concentration increases until conditions for a combustion are met and hydrogen combustion occurs. The operating deck and secondary containment are assumed fully damaged in this case.

The following mechanisms are addressed for the SFP. These are mentioned in the EPRI report [Ref-25.82] or NUREG-1738 [Ref-25.88] as areas of uncertainty relevant to the progression of severe accidents in the SFP.

- Zircaloy fire

  In the EPRI SFP PSA report [Ref-25.82], a Zircaloy fire is discussed:

  > *A Zircaloy fire in the SFP may only occur due to a significant LOCA event such that the SFP is drained below the bottom of the fuel racks prior to fuel heatup and melt. Experimental tests for Zircaloy fires indicate that significant air flow is required up through the bottom of the air channels to support initiation of a Zircaloy fire. Without the air flow, which would be significantly precluded if water covers the bottom of the fuel racks, the probability for a Zircaloy fire is significantly reduced. In addition, for slow SFP boildown events, fuel melt is*

expected to occur prior to potential evaporation of the SFP inventory. For these cases, the fuel is judged to relocate such that the inadequate air flow through the fuel racks would make the probability of a Zircaloy fire very low. Therefore, Zircaloy fires are currently judged to be limited to significant LOCA events that uncover the bottom of the fuel racks and not to apply to SFP boildown type events or other LOCA events.

- Re-criticality

The re-criticality issue is discussed in NUREG-1738 [Ref-25.88] and the EPRI SFP PSA report [Ref-25.82]:

Potential criticality events, such as events involving loose pellets or the impact of water (adding neutron moderation) during personnel actions in response to accidents, were discounted because the basic physics and neutronic properties of the racks and fuel would prevent criticality conditions from being reached with any credible likelihood. For example, without moderation fuel at current enrichment limits (no greater than 5 wt% U-235) cannot achieve criticality, no matter what the configuration. If it is assumed that the pool water is lost, a reflooding of the storage racks with unborated water may occur during personnel actions. However, BWR storage racks are designed to remain subcritical if moderated by unborated water in the normal configuration. Thus, the only potential credible scenarios are the two scenarios described above, which involve crushing of fuel assemblies in low-density racks or degradation of Boraflex over long periods in time. These conclusions assume present lightwater uranium oxide reactor fuel designs. Alternative fuel designs, such as mixed oxide (MOX) fuels will have to be reassessed to ensure that additional vulnerabilities for pool criticality do not exist. [Ref-25.88]

During reflood of a drained pool for a region of the pool storing higher reactivity fuel assemblies re-criticality is postulated as possible. If such an event affected a region of the pool (as opposed to only a portion of a particular assembly), and if it occurred at a point in the accident where the fuel was only partially covered, the event could have an impact on onsite dose rates. [Ref-25.82]

- Core–concrete interactions

Core-concrete interaction is discussed in the EPRI SFP PSA report [Ref-25.82]:

If SFP cooling becomes unavailable, the hottest bundles may undergo clad failure and fuel melting. The next phase of the severe accident progression would involve molten debris candling on the fuel bundles and eventually falling to the floor of the SFP. If there is residual water in the SFP there may be an extended period of debris cooling (steaming) from the pool. Once there is dryout of the debris, then the steel liner would be eviscerated and the debris would initiate core concrete interactions that would result in exothermic chemical reactions with the release of Carbon Monoxide (CO), which is a combustible gas.

Progression of the core-concrete interaction depends on the heat flux of debris. The decay heat level of spent fuel in the SFP is low because the cooling time of the fuel is long. Therefore, the core-concrete interaction in the SFP is not considered.

- Air cooled fuel assembly

Air cooling of fuel assemblies is discussed in the EPRI SFP PSA report [Ref-25.82]:

There are several heat transfer mechanisms that can enter into the cooling of spent fuel during various postulated severe accident scenarios. These include air cooling of the bundles. Convective air cooling is a cooling mechanism of last resort in which the complete loss of water from the SFP opens up possible convective flow paths that are buoyancy driven and *create high velocity upward air flow through the interior of canisters and draw air down on the outside of the canisters.*

This cooling method is of course very sensitive to:

- Lattice design

- Density of fuel assemblies

- Orifices at the bottom of the canisters

- Blockage of the bottom of the canisters (e.g., by water or debris in the SFP)

Cooling processes such as air cooling within the SFP are expected to be shown to be effective, as indicated by Sandia testing and MELCOR calculations, but these are yet to be confirmed for realistic fuel checkerboard SFP loadings.

As mentioned above, fuel assembly coolability by air has large uncertainties. Therefore, air cooling of spent fuel is not credited in the SFP Level 2 PSA.

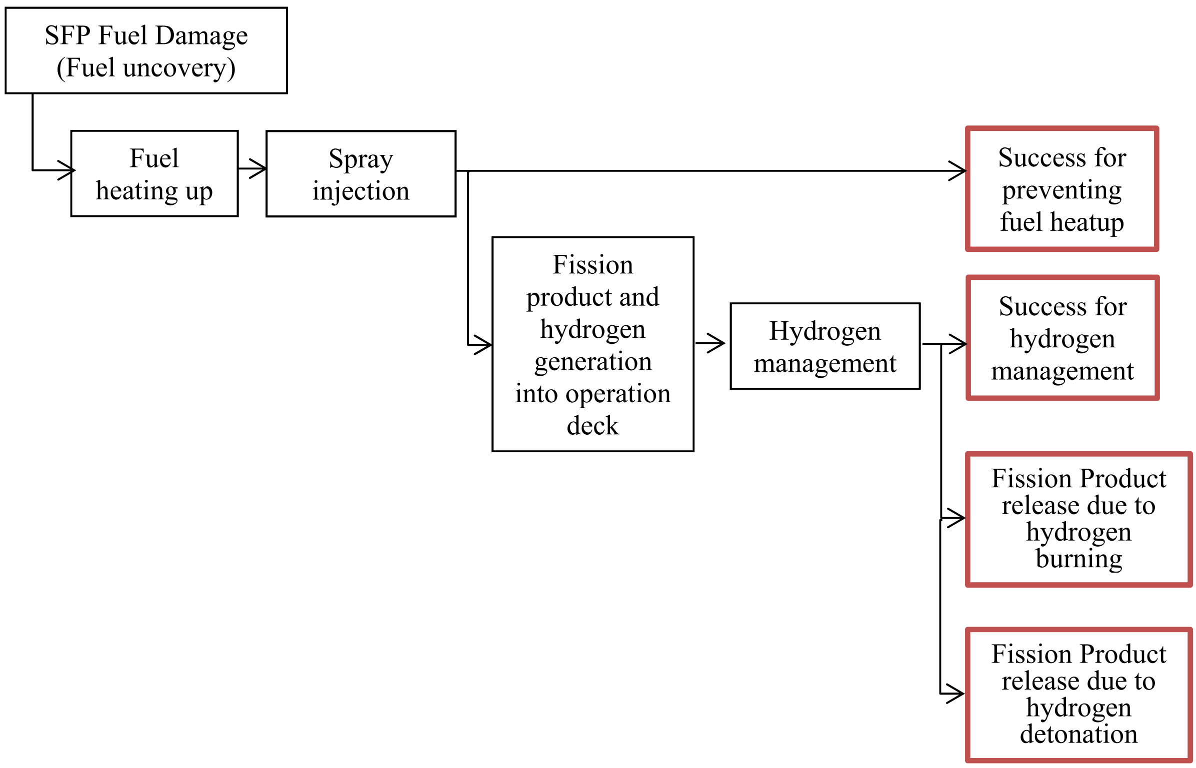

Figure 25.9.6-2 Outline of Accident Progression for SFP

Severe Accident Progression Analysis

(1)Purpose of Severe Accident Progression Analysis

Accident progression analyses of each PDS were performed in order to obtain the data which is necessary for the development of the containment event tree such as plant hydrological behaviour, chronology of accident progression (fuel heatup after the water level reaching TAF), and the fission product release.

(2) Evaluation Method for Severe Accident Progression Analysis

The MAAP 5.03 code was used for accident progression analysis in Level 2 PSA of the UK ABWR for the SFP. This analysis included models for the important accident phenomena that might occur in the SFP and in the R/B. The MAAP code calculated the progression of the postulated accident sequence, including the deposition of the fission products, from initiating events to either a safe, stable state or to an impaired SFP and R/B opened condition (due to steaming or hydrogen effects). The MAAP code also calculated fission product releases to the environment.

(3) Accident Progression Analysis Condition

Table 25.9.6-1 shows the summary of analysis conditions. Analysis conditions for all accident sequences are indicated in Table 25.9.6-1. The nodalisation of spent fuel, the SFP and the R/B are shown in [Ref-25.76].

In this analysis, none of the potential mitigation measures were considered because the analyses were conducted to obtain the source term and time to actual fuel damage (fuel heatup) for the PSA.

Table 25.9.6-1 UKABWR SFP Analysis Conditions (1/2)

Items	Conditions	Remarks
Initial Decay Heat	-	The decay heat is conservatively assumed to be value of 4 days after shutdown. It corresponds to the decay heat power at the beginning of POS B and it gives the highest decay heat power for hot fuel bundles.
Initial Water Level of SFP	a. 0.0 m (Catastrophic failure) b. 11.51 m (Boil-off or LOCA condition)	11.51 m：Normal level
SFP Atmosphere	Air	Refueling floor is not inerted.
Free Volume of SFP	2,195 m^3	The design of SFP volume is under development. This is based on the value of reference ABWR plant.
Free Volume of R/B Decay Heat Power	36,743 m^3 ANSI/ANS-5.1-1979 or NRC REGULATORY GUIDE 3.54	This is design value of UK ABWR. Volume of the refueling floor is considered. MAAP503 decay power model consists of 2 formulas. ANSI/AN-1979 for Cooling time ＜ 1 year, US NRC REGULATORY GUIDE 3.54 for ≧ 1 year.
Fuel Type	GE-14	10x10 fuel rods
Number of Fuel Nodes	Number of racks: 15 Axial Nodes : 27 nodes	Although there are 31 racks in the spent fuel pool, only 15 racks are used. It leads to high power density in the racks and gives conservative condition for severe accident analysis.
Number of SFP bundles	1,744 Bundles (200 percent Core) Hot bundles: 225 Cold bundles : 1,519	
Initial Pressure	Atmospheric pressure	Environmental pressure is assumed.
Initial Gas Temperature	300 K	Environmental temperature is assumed.
Initial Pool Temperature	325 K	The temperature of the SFP water is normally maintained below 52 ^{o}C. (325 K)

Table 25.9.6-1 UKABWR SFP Analysis Conditions (2/2)

Items	Conditions	Remarks
LOCA status	Boil-off / LOCA / Catastrophic failure	Each status is considered.
SFP spray status	Success / Fail	The design of SFP spray is under development. Both statuses are considered in catastrophic failure scenario.
SFP spray flow rate	120 m^3/h	This flow rate is described in the Accident Management Facility Basic Requirement Specification. SFP spray is assumed to distribute uniformly in the whole area of the SFP.
SFP spray droplet size	2 mm	Large droplet size in the spray experiment is assumed. This is conservative setting because DF of the SFP spray decreases as increasing the droplet size.
SGTS status	Fail	SGTS is not designed to filter the fission products generated in a severe accident. Therefore, SGTS operation is not assumed to operate in the severe accident analysis.
Blowout panel status	Success	If the blowout panel does not open successfully, the pressure in the R/B increases and the blowout panel finally opens because the blowout panel is the weakest point in the R/B due to overpressure. Therefore, the blowout panel is assumed to open in the severe accident analysis.
PARs in the refueling floor	Not Considered	As the design of PARs is under development, PARs are not considered in the severe accident analysis.

(4) Accident Progression Analysis Results

Time to TAF/Fuel Damage

Time to TAF/fuel damage taken from the accident progression analysis is given in Table 25.9.6-2. Time to TAF is not exactly the same as in the SFP Level 1 PSA because of the initial condition of the SFP water and spent fuel decay heat.

Table 25.9.6-2 Time to TAF/Fuel Damage

Case No.	LOCA Status	SFP Spray Status after Fuel Damage	Decay heat status[4]	Time to TAF (h)	Time to Fuel Damage (h)
Case 1	Boil-off	Fail	4 days after shutdown	352.0	407.6
Case 2	LOCA[1]	Fail	4 days after shutdown	62.2	75.5
Case 3	Catastrophic failure of SFP[2]	Fail	4 days after shutdown	0.0	0/8
Case 4		Success[3]	4 days after shutdown	0.0	0.8
Case 5			16 days after shutdown	0.0	-

(i) LOCA flow rate in SFP is assumed to be 30 m^3/h at normal water level.

(ii) No water is assumed to be in SFP from the beginning of an accident. Blow-out panel is assumed to open at the beginning of an accident because of the following:

- Integrity of the R/B cannot be taken credit in case of catastrophic failure.
- Fission Product mass fraction released to environment increases when blowout panel is opened. It gives conservative result in relation to the source term assessment.

(iii) SFP spray flow rate is assumed to 120 m^3/h. SFP spray is assumed to be initiated 0.5 hours after an initiating event.

(iv) The purpose of this source term analysis is to get bounding result of fission product release and time available before the fuel heatup. Therefore, the analysis condition of decay heat is conservatively set to 4 days after shutdown. On the other hand, the spent fuel is moved to SFP and then the SFP gate is closed at least 16 days after shutdown including the case of full core off-load (i.e. POS E). The timing of the SFP gate closing (16 days after shutdown) is consistent with the SFP PSA model. Therefore, the representative condition for SFP is “16 days after shutdown” and the case 5 is adopted.

Spent fuel heatup

After the water level decreases below the TAF, spent fuel temperature increases (heatup). This heatup rate does not depend on the total decay heat in the SFP but the decay heat of individual single fuel assembly. Figure 25.9.6-4 and Figure 25.9.6-5 show the Case 1 (Boil-off) and Case 2 (drain down (small leak)) heatup behaviour from the start of heatup. Accident progression analysis by MAAP uses the fuel damage definition as 700 °C fuel temperature. However, NUREG-1738 [Ref-25.88] stated the following.

For assessing the onset of fission product release under transient conditions (to establish the critical decay time for determining availability of 10 hours to evacuate) it is acceptable to use a temperature of 900°C if fuel and cladding oxidation occur in air. If steam kinetics dominate the transient heatup case, as it would in many boildown and draindown scenarios, then a suitable *temperature criterion would be around 1,200 °C. For establishing long-term equilibrium conditions for fuel pool integrity during SFP accidents which preclude significant fission product release it is necessary to limit temperatures to values of 600 °C to 800 °C.. If the critical decay time is sufficiently long (> 5 yrs) that ruthenium inventories have substantially decayed then it would be appropriate to consider the use of a higher temperature, 800 °C, otherwise fission product releases should be assumed to commence at 600 °C. These cases are marked by substantial time for corrective action to restore cooling and prevent smaller gap type releases associated with early cladding failures.*

This analysis is not exactly the same condition as in the Level 1 UK ABWR SFP PSA but the acceptance criteria to prevent the large fission product release from heatup of spent fuel is conservatively assumed as 600 °C using the above information.

According to Figure 25.9.6-4 and Figure 25.9.6-5 , additional time margin in the SFP Level 2 PSA (to prevent the large fission product release from heatup spent fuel) based on 600 °C assumption are described as follows.

- PDS: Boil-off

  27 hours

- PDS: Drain down (small leak)

  5 hours

- PDS: Drain down (large leak)

  0.5 hour

For the PDS: Drain down (large leak), effectiveness of fuel cooling is directly derived from the case 5 analysis result of accident progression analysis because decay heat level of the SFP PSA is equal or less than decay heat level at 16 days after the reactor shutdown.

Source Term

Fission product release from the SFP for each PDS was assessed based on the accident progression analysis. Table 25.9.6-3 shows the source term results.

The release category is discussed in Section 25.9.6.8.

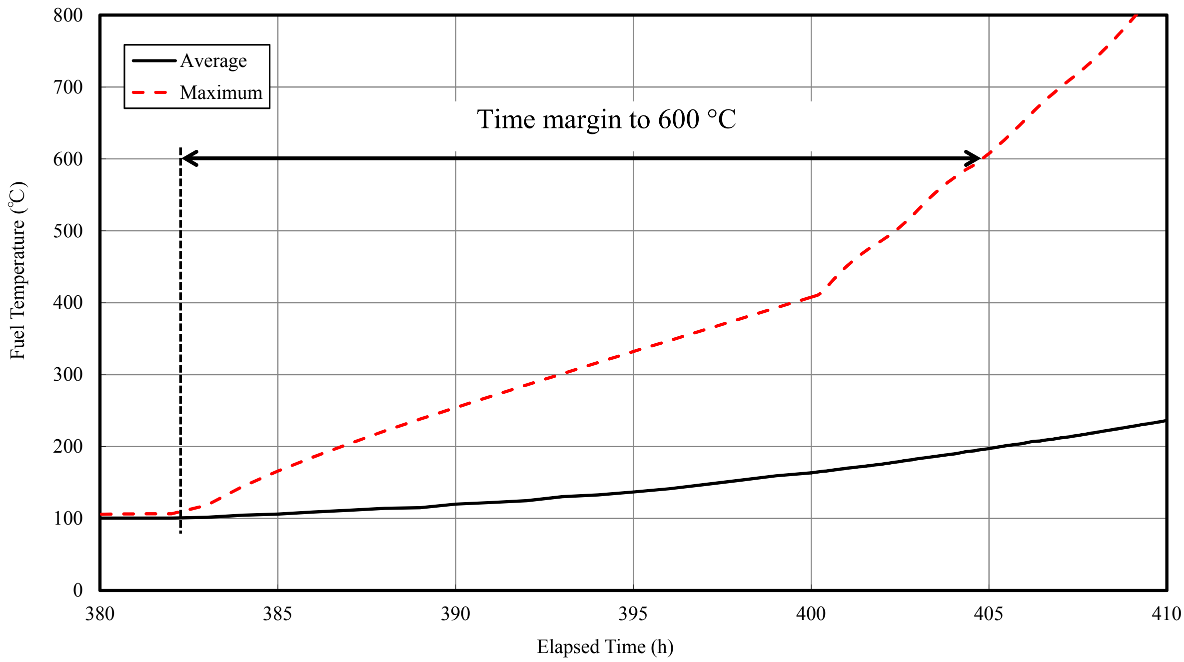

Figure 25.9.6-4 Fuel Heatup Behaviour in Boil off, 4 Days Decay Heat after Shutdown

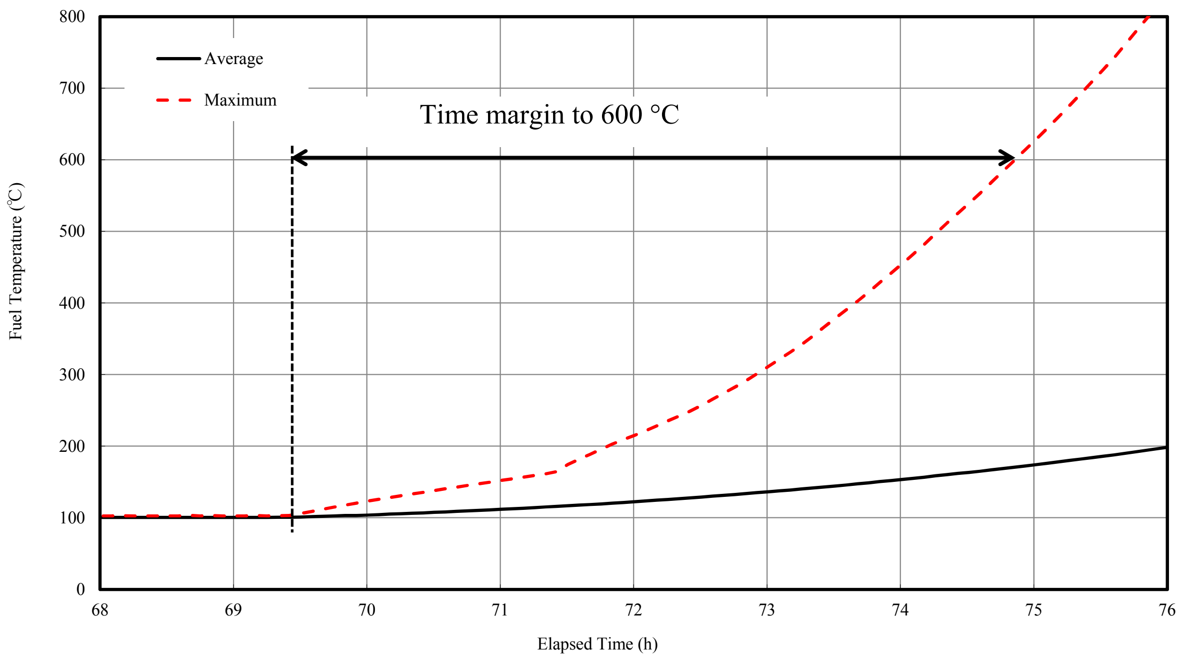

Figure 25.9.6-5 Fuel Heatup Behaviour in SFP Failure, 4 Days Decay Heat after Shutdown

Table 25.9.6-3 Source Term of Release Categories in SFP

Case No.	Time to TAF [hr]	Time to Fuel Damage [hr]	Time to Hydrogen Burn [hr]	FP Mass Fraction Released to Environment											
				Noble Gas	CsI	TeO_2	SrO	MoO_2	CsOH	BaO	La_2O_3	CeO_2	Sb	Te_2	UO_2
Case 1	352.0 hr	407.6 hr	-	9.58E-01	9.38E-01	8.76E-01	6.73E-01	7.61E-01	9.38E-01	7.17E-01	2.64E-01	3.04E-01	8.48E-01	0.00E+00	0.00E+00
Case 2	62.2 hr	75.5 hr	-	9.46E-01	8.15E-01	7.51E-01	6.16E-01	6.24E-01	8.15E-01	6.37E-01	1.05E-01	2.31E-01	7.30E-01	0.00E+00	0.00E+00
Case 3	0.0 hr	0.8 hr	-	8.01E-01	7.25E-01	3.87E-01	8.92E-02	1.24E-05	7.25E-01	1.44E-01	1.62E-03	2.90E-03	3.76E-01	0.00E+00	0.00E+00
Case 4	0.0 hr	0.8 hr	-	8.01E-01	6.35E-01	3.01E-01	2.80E-02	3.91E-05	6.35E-01	4.92E-02	3.82E-04	1.79E-03	2.65E-01	0.00E+00	0.00E+00
Case 5	0.0 hr	-	-	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

25.9.6.5 Containment Performance Analysis

A core damage event resulting in containment failure followed by failure of long term heat removal from containment may degrade the R/B environment. These effects are considered as initiating event in this SFP PSA as "reactor challenge" as mentioned in 9.2.2 (4).

In this initiating event, SFP structural integrity including that of the liner is assumed to be lost due to hydrogen burning effects or strain of SFP structural material by RCCV pressure with an assumed conditional probability. Different conditional probabilities are assumed for each containment failure mode.

In addition, SFP cooling or support systems in the R/B such as FPC, RHR, RCW, emergency AC power supply and so on are assumed to fail due to steaming in the R/B, but the makeup function from sources outside the R/B (FLSS and FLSR) can be credited.

Moreover, some containment failure sequences with no core damage also release steam into the R/B. SFP structural integrity including that of the liner assumed to be lost due to strain of the SFP structural material by RCCV overpressure with an assumed conditional probability. SFP cooling or support systems in the R/B such as FPC, RHR, RCW, emergency AC power supply etc. are assumed to fail but the makeup function from sources outside the R/B (FLSS and FLSR) can be credited.

The conditional probability for each effect on the SFP was set. This probability is based on the "Spent Fuel Pool Risk Assessment Integration Framework (Mark I and II BWRs) and Pilot Plant Application" [Ref-25.82] and discussion of its applicability to the UK ABWR with the containment design engineer and the SFP design engineer.

25.9.6.6 Secondary Containment Performance Analysis and Mitigation Systems

Since the SFP is located outside of primary containment, fission products are initially released to the operating deck in the fuel damage sequences for the SFP. The operating deck is a part of R/B and secondary containment.

Secondary containment is provided by the boundary of the R/B, its HVAC systems, and SGTS. In addition, a blowout panel is installed in the R/B. The boundary is maintained regardless of plant operational states such as at Power condition or shutdown condition:

- Personnel access (three locations) and material entrance (one truck access location) to the secondary containment consist of airlocks with interlocked double doors or hatches. Both double doors and hatches are not opened at the same time.
- All the penetrations are sufficiently sealed providing a high degree of air tightness in the secondary containment.
- Upon receiving emergency signal, the reactor area (R/A) HVAC is automatically shut down and the R/A Isolation Dampers mounted on the Supply/Exhaust duct are automatically closed to isolate the R/A and prevent exfiltration of the radioactive gas to outside by switching to the SGTS.

In the following two cases, the boundary function is lost.

- The blowout panel opens due to high energy pipe break accident or hydrogen management
- Station blackout which causes loss of interlock, HVAC and SGTS

25.9.6.7 Containment Event Trees

CET Entries

The generic CET for the SFP Level 2 PSA is shown as Figure 25.9.6-6.

This CET is applied to each PDS defined. Headings (top events) for the CET consider each mitigation system. End states are defined from this CET as release categories for each PDS.

PDS	SFP SPRAY	RB VENT	HBURN	Release category
PDS	SFP Spray after uncovery	Reactor building venting (BOP opening)	Hydrogen burning	

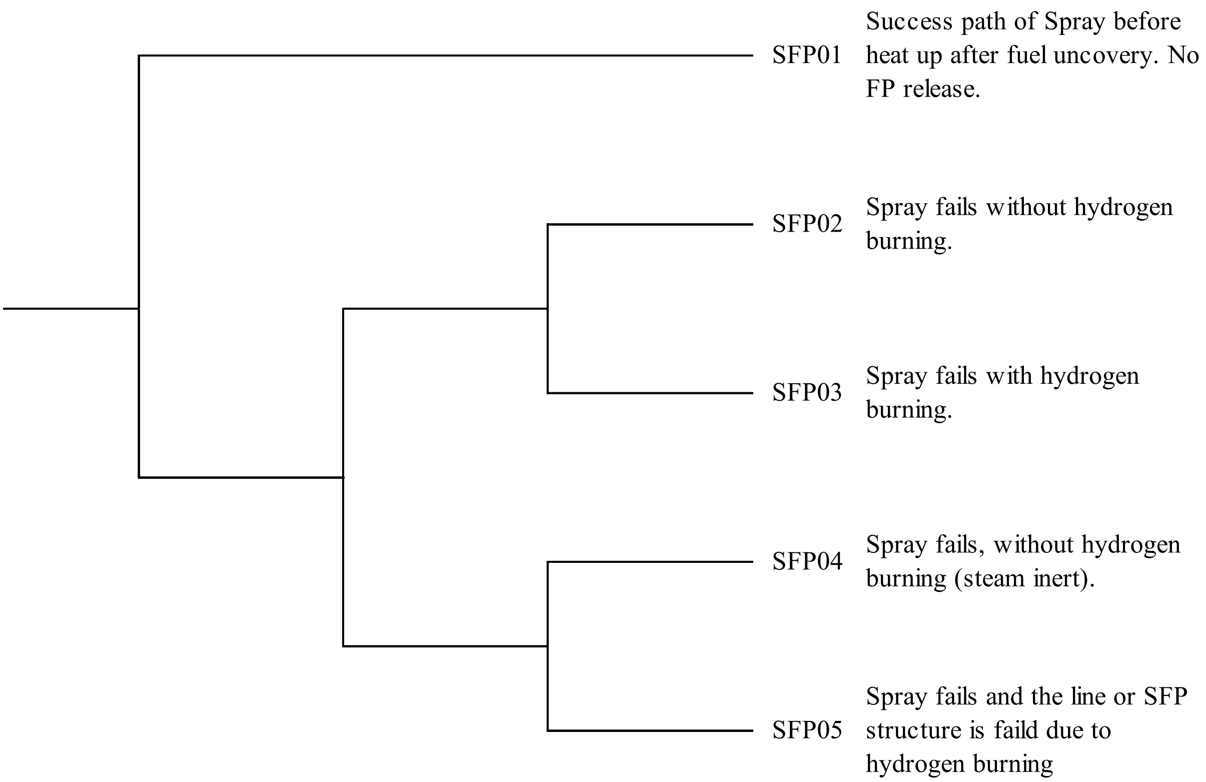

Figure 25.9.6-6 Generic CET for SFP PSA

Mitigating Systems

(1) SFP spray

FLSS and FLSR both provide a mitigation function by spraying water onto spent fuel for cooling and scrubbing of fission products in the case of some fuel uncovery sequence [Ref-25.84]. The safety function, system description and additional support information are described below.

Safety function

- Fuel cooling

  Fission product release from spent fuel can be prevented by spray injection within appropriate time margin (time to uncovery to fuel heatup) from FLSS or FLSR.

- Scrubbing effect

  Scrubbing effect for fission product release is not credited in the SFP Level 2 PSA.

  Scrubbing effect calculation results by using MAAP significantly depend on droplet size (diameter) of spray [Ref-25.82]. Correlation between spray droplet size and scrubbing effect is not verified by appropriate benchmarking. Although EPRI's "SFP Risk Assessment Integration Framework (Mark I and II BWRs) and Pilot Plant Application" credited the scrubbing effect (mitigate fission product release to Medium range) as an assumption, source term calculations for the UK ABWR do not credit the scrubbing effect.

- FLSS/FLSR system description

  Injection from both FLSS and FLSR use the spray nozzles and there are no switching lines.

- Instrumentation for SFP severe accident (SA)

  In the beyond design basis scenario such as significantly low SFP water level (e.g. around spent fuel TAF), dedicated instrumentation (water level and water temperature monitor) for such severe accident scenario is planned to be installed in SFP.

  Water level monitoring in SA uses multiple thermocouples and has a switch-on lamp as an indicator in both the MCR and backup building [Ref-25.89]. An alarm is assumed to be installed with this SA water level monitoring when the SFP water level decreased to around TAF, which is the "cue" to operator manual action for the SFP spray after fuel uncovery. This instrumentation is considered as a hardware failure affecting human failure events for SFP Level 2 PSA. Because of this additional cue in Level 2 PSA phase, complete dependency between Level 1 PSA and Level 2 PSA is not applied to manual spray operation.

(2) Hydrogen management

Flammable gas (hydrogen) management is prepared for the secondary containment. Countermeasures for hydrogen release are summarised;

- Small amounts of hydrogen leakage to the R/B SGTS
- Large amount of hydrogen leakage to the R/B are mitigated by R/B Opening with Defense Shield
- Further reduction of hydrogen concentration to the R/B is possible using PAR

SGTS:

SGTS can treat a limited volume of hydrogen. In addition, SGTS cannot be credited when Class 1 AC is not available.

Passive Auto-catalytic Recombiner (PAR):

PAR is a device that recombines the hydrogen gas and oxygen gas without a need for external power or operator action. PARs are already being implemented in some plants (e.g. in Belgium) as a means to improve safety margins for hydrogen combustion in severe accident situations. The installation of PARs is highly influenced by geometric and operational constraints (access to maintenance areas should remain free; PARs must be accessible for periodic surveillance, etc.). Therefore, locations of PARs are optimised at the design stage. Calculations and experiments showed that the exact location of a PAR in a compartment was not critical for its performance. The strong convection created by a working PAR effectively mixes the atmosphere.

Blow out panel:

Generally, blowout panels are installed to mitigate the pressurisation of the R/B (secondary containment) due to steam from a break outside containment (BOC) or ISLOCA. In addition, the blowout panel is opened to prevent hydrogen burning when a severe accident occurs and hydrogen leaks from containment. The latter function is credited in the SFP Level 2 PSA.

A blowout panel is opened by the operating deck pressurisation at 3.43 [kPa (gauge)] or by manual operator actions. If a BOC or ISLOCA occurs, pressure of the R/B increases rapidly and then the blowout panel is opened. The pressure increase due to steam from SFP decay heat is slow. In this condition, the blowout panel assumed to be not opened because the steam leaks within the design leak flow rate of SGTS boundary. Operator actions for blowout panel opening are also assumed to fail because the operator cannot access the operating deck due to R/B steaming or high radiation dose due to loss of shielding by SFP water.

(3) CET Top Events

CET top events are listed as following.

Top events for SFP PSA CET

No.	Top event	Outline
1	SFP SPRAY	SFP Spray after uncovery
2	RB VENT	R/B venting (BOP opening)
3	HBURN	Hydrogen burning

Top event: SFP SPRAY

This top event denotes the spray function to SFP to prevent fission product release.

For the PDS: boil-off and drain down (small leak), FLSS injection has already failed in Level 1 PSA phase. The significant cutsets in Level 1 PSA include HFE “failure of manual initiation of FLSS” and it has limited dependency to Level 2 PSA because of SA dedicated instrumentation and alarm. Therefore, the operator can try to use FLSS again cued by the alarm after fuel uncovery.

FLSR also has a spray injection function but operator will have difficulty accessing the R/B to connect FLSR due to significant steaming in R/B. Therefore, FLSR is not credited.

In the PDS: drain down (small leak), SFP leak due to hydrogen combustion based on IEAP Level 2 PSA results (reactor challenge) sequence is included. In this sequence, piping of FLSS and FLSR are assumed to be damaged by hydrogen burning. Therefore, the spray function by FLSS is not credited.

Top event: RB VENT

This top event denotes the blowout panel opening. There are a number of hydrogen management measures in secondary containment (operating deck) but the blowout panel is only credited in this SFP Level 2 PSA and this system can be credited in BOC/ISLOCA (initiating event ID: SFP_LOCA_F). The failure probability is represented by a check valve fail to open because of similarity of the failure mechanism.

In the PDS: drain down (small leak), SFP leak due to hydrogen combustion based on IEAP Level 2 PSA results (reactor challenge) sequence is included. In this sequence, hydrogen burning has already occurred, and then RB VENT is not effective. Therefore, the blowout panel opening is not credited in the sequence.

Top event: HBURN

The possibility of hydrogen burning is not completely rejected even if the blowout panel is opened. If hydrogen burning occurs in spite of success of the blowout panel opening, the boundary of secondary containment will be damaged and fission products will be release to environment.

RB VENT success case: Fuel damage sequence due to reactor challenge “LOCA (BOC/ISLOCA)” can get this path. In this case, a large amount of steam is generated by the SFP and the operating deck is inerted. In addition, the blowout panel opening can vent the hydrogen. Probability of hydrogen burning is assumed to be 0.01 as mean value based on NUREG/CR-4700 “very unlikely” [Ref-25.50]. For this probability, a beta distribution and alpha factor of 1 are assumed.

RB VENT failure case: In this case, the blowout panel is not opened, and then venting effect is not available. However, since a large amount of steam is generated by the SFP, the operating deck is inerted. In the PDS: drain down (large leak), this PDS does not involve a total loss of SFP water. Some water may remain under TAF in the SFP. Therefore, steam inerting is credited. Probability of hydrogen burning is assumed to be 0.1 as mean value based on NUREG/CR-4700 “unlikely” [Ref-25.50]. For this probability, beta distribution and alpha factor of 0.3 are assumed. Since this allocated probability is too high to maintain the rare event approximation applied to frequency evaluation for HBURN success scenario, success probability 0.9 is used for HBURN success scenario.

Dependencies for CET top events

Table 25.9.6-4 shows dependency matrix of the headings in the containment event trees of the SFP Level 2 PSA. “X” in the table denotes total dependency. “XX” denotes the partial dependency. These are explained as below.

- Dependency between “SFP SPRAY”, “RB VENT” and “HBURN”

  If SFP SPRAY succeeds, the spent fuels are successfully cooled and the accident progression stops. Therefore, “SFP SPRAY” affects “RB VENT” and “HBURN” and these headings have total dependency.

- Dependency between “RB VENT” and “HBURN”

  If RB VENT is failed, hydrogen concentration in the R/B becomes high. Therefore, “RB VENT” affects the probability of “HBURN”, and these headings have partial dependency.

- Dependency associated with CCF: These three top events have no dependency associated with CCF of equipment, support systems and operator actions.

  - SFP spray denotes spray function from FLSS. RB venting denotes the blowout panel opening. HBURN denotes the hydrogen combustion phenomena. These three top events have no common equipment. Therefore, there is no dependency due to CCF.

- In the three top events, support system is needed only SFP SPRAY. Therefore, there is no dependency due to support system failure.
- In these three top events, operator action is credited only SFP SPRAY. Therefore, there is no dependency due to operator action dependency.

Table 25.9.6-4 Dependency Matrix of the Headings in the Containment Event Trees

	SFP SPRAY	RB VENT	HBURN
SFP SPRAY			
RB VENT	X		
HBURN	X	XX	

Accident Sequence Mission Time

In order to determine that a sequence is a success or failure, a mission time must be assumed for analysis. The mission time is defined as the time period that a system or component is required to operate in order to successfully perform its function. The typical plant mission time used in Level 1 PSAs is 24 hours for sequences which result in a safe stable state. The assumption of 24 hours is based on several factors including:

- Simplifying assumptions made in PSA analysis, no credit for repair of failed equipment after initiating event and credit for operator actions limited to existing approved, procedures are overly conservative for times greater than 24 hours.
- The resources available to the operator, including offsite resources, when responding to an event past 24 hours, cannot be predicted.
- Emergency response actions external to the plant, such as public evacuations, are more reliable at time frames greater than 24 hours.

The exceptions include systems and components which are modelled with functions lasting less than the plant mission time, batteries in loss of offsite power events which are recovered in a given time frame. In addition, longer time mission times have been used in the analysis of passive systems which rely in finite sources such as cooling water.

In the Level 2 PSA analyses, a typical equipment mission time is 24 hours after the onset of fuel damage (fuel uncovery). The bases for this assumption are the same as described for the Level 1 PSA analysis (section 25.4). The mission time to determine the magnitude and characteristics of the fission product releases should indicate that the fission product release has plateaued or levelled off. Therefore, the Level 2 PSA mission time for fission product release determination should be longer than the equipment mission time.

For the UK ABWR Level 2 PSA, the mission time to determine the magnitude and characteristics of the fission product releases is 36 hours after the onset of fuel damage.

Release Categories

The three PDS and the generic CET result in the following end states in SFP Level 2 PSA. Each end states are defined as release category and these are explained as follows.

Release category for PDS: boil-off (BO-SFP)
BO-SFP01: In this release category, fuel heatup is terminated by spray after fuel uncovery.

BO-SFP02: In this release category, spray has failed and the blowout panel opening has also failed. However, hydrogen burning has not occurred. This RC categorised into large release in late phase.

BO-SFP03: In this release category, spray has failed and blowout panel opening has also failed. Finally, hydrogen burning occurs. This release category is categorised into large release in late phase.

Release category for PDS: boil-off due to BOC/ISLOCA (BO-SFP_LOCA)

BO-SFP_LOCA 01: In this release category, fuel heatup is terminated by spray after fuel uncovery.

BO-SFP_LOCA 02: In this release category, spray has failed but the blowout panel opening is successful and hydrogen burning does not occur. This release category is categorised into large release in late phase.

BO-SFP_LOCA 03: In this release category, spray fails. Blowout panel opening is successful but hydrogen burning occurs. This release category is categorised into large release in late phase.

BO-SFP_LOCA 04: In this release category, spray has failed and the blowout panel opening is also failed. However, hydrogen burning does not occur. This release category is categorised into large release in late phase.

BO-SFP_LOCA 05: In this release category, spray has failed and the blowout panel opening has also failed. Finally, hydrogen burning occurs. This release category is categorised into large release in late phase.

Release category for PDS: drain down (small leak)

DDS-SFP01: In this release category, fuel heatup is terminated by spray after fuel uncovery.

DDS-SFP02: In this release category, spray is failed but blowout panel opening is succeeded and hydrogen burning is not occurred. This RC categorised into large release in late phase.

DDS-SFP03: In this release category, spray fails. The blowout panel opening succeeds but hydrogen burning does occur. This release category is categorised into large release in late phase.

DDS-SFP04: In this release category, spray fails and blowout panel opening also fails. However, hydrogen burning does not occur. This release category is categorised into large release in late phase.

DDS-SFP05: In this release category, spray fails and blowout panel opening also fails. Finally, hydrogen burning occurs. This release category is categorised into large release in late phase.

Release category for PDS: drain down (large leak)

DDL-SFP01: In this release category, fuel heatup is terminated by spray after fuel uncovery.

DDL-SFP02: In this release category, spray fails and blowout panel opening also fails. However, hydrogen burning does not occur. This release category is categorised into large release in early phase.

DDL-SFP03: In this release category, spray fails and blowout panel opening also fails. Finally, hydrogen burning occurs. This release category is categorised into large release in early phase.

Source Term Analysis results

Source term analysis conditions for the release categories are described in Table 25.9.6-1.

The fission product aerosol removal processes which are deposition, impaction and thermophoresis are considered in all compartments including the R/B. The R/B is modelled using one compartment and inner rooms and inside walls are not included.

The source term cases are analysed to more than 300 hours, which in all cases were > 36 hours after core damage. Based on the trend of fission product release, if the termination time was extended, the result of source term would not change.

25.9.7 Results (SFP Level 2)

The purpose of this section is to document the calculation of the LRF/LERF due to events in all SFP fuel damage sequence, including full power and shutdown*.

*NOTICE: As mentioned in section 25.9.3.1, accident class VIII in shutdown PSA POS-B is included in SFP Level 2 PSA Scope. POS-B accident class VI is not included in SFP Level 2 PSA scope and then probabilistically addressed in shutdown Level 2 PSA. The source term analysis is performed in SFP Level 2 PSA and the information is shared with shutdown Level 2 PSA.

25.9.7.1 Model Results Summary

The UK ABWR PSA model consists of containment event trees and fault trees that are quantified using a fault tree linking process. The calculation of the total LRF is performed as a single top gate.

As a result of quantification, the total LRF resulting from a single top gate is 2.84E-08 /y when a truncation value of 1.0E-14 /y is used for quantification. Details of the results are described in the next subsections.

(a) Significant POS

Figure 25.9.7-1 shows a summary contribution in the form of a pie chart. LRFs for POSs are shown below.

- POS S : 1.24E-09 /y
- POS A : 2.35E-08 /y
- POS B1 : 1.55E-10 /y
- POS B2 : 8.54E-10 /y
- POS C : 3.61E-09 /y
- POS D : 1.50E-08 /y
- POS E : 5.42E-10 /y
- POS F : 2.95E-09 /y

POS A has the highest FDF (about 49 percent of the total LRF). The initiating event “Loss of Class 1 AC due to human error” in POS A contributes to about 41 percent of the total LRF.

POS D has the second highest LRF (about 31 percent of the total LRF). During this POS, only 1 FLSS train is available due to maintenance schedule.

The third contributor is POS C (about 8 percent of the total LRF). In POS C, only division 2 is available. Division 1 and 3 are out of service and division 3 effectiveness is limited. Especially for the POS C, the number of credited systems is limited and then FLSS importance is relatively high from the view point of shutdown Level 1 PSA accident sequence. In this case, not only human failure events but also mechanical failures have significant contributions. In the Level 2 PSA treatment, the effect of SFP spray is limited because the FLSS completely depends on its mechanical failure in Level 1 PSA cutset.

Despite the highest decay heat, POS S has the small LRF since it has the most redundant mitigation systems and the shortest duration.

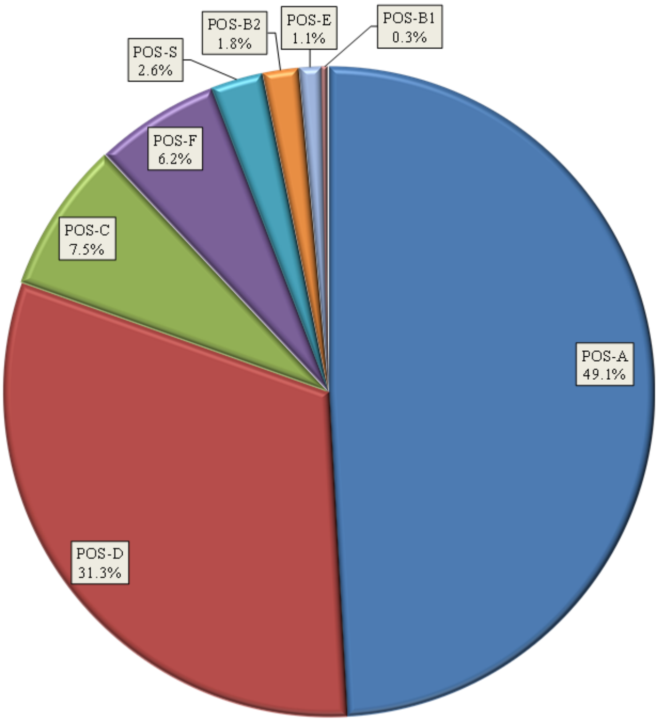

Figure 25.9.7-1 Contribution to LRF by POS

(b) Significant Plant damage states

LRFs for three defined three PDSs are shown in Table 25.9.7-1.

- Boil-off : 4.77E-08 /y
- Drain down (large leak) : 8.85E-11 /y
- Drain down (small leak) : 1.10E-11 /y

Table 25.9.7-1 Relationship between Level1 and Level2 Result

	Level 1		**Level 2**
Boil-off	FDF in Level 1 SFP PSA for boil-off (4.2E-07)	Total 4.6E-07 /y	4.77E-08 /y
	FDF in Level 1 Shutdown PSA for Accident class VIII (1.8E-08)		
	Loss of SFP cooling due to R/B steaming (1.7E-08)		
Drain down (large leak)	FDF in the Level 1 SFP PSA for drain down (large) due to strain or deformation (reactor challenge) (1.5E-09)	1.5E-09 /y	8.85E-11 /y
Drain down (small leak)	SFP Leak due to hydrogen burning (reactor challenge) (1.1E-11)	1.11E-11 /y	1.10E-11 /y

PDS Boil-off has the highest LRF (about 99.8 percent of the total LRF). Summation of the input for this PDS, i.e., FDF in the Level 1 SFP PSA for boil off, FDF in Level 1 shutdown PSA for accident class VIII (eight) and loss of SFP cooling due to R/B steaming (reactor challenge), is 4.6E-07. LRF for this PDS is about 10 percent of the input frequency. This is because a large part of the cutsets for this PDS includes the human failure event for FLSS and it can be credited again in Level 2 PSA in SFP_SPRAY heading. Success of SFP SPRAY prevents the fuel heating up and large release.

The second contributor is PDS Drain down (large leak) (about 0.2 percent of the total LRF). Summation of the input for this PDS i.e., FDF in the Level 1 SFP PSA for drain down (large) due to strain or deformation (reactor challenge) is 1.5E-09. LRF for this PDS is about 6 percent of the input frequency. No cutsets for this PDS include the failure event related to FLSS because FLSS is not credited in the Level 1 PSA. Thus, FLSS can be credited in Level 2 PSA in SFP_SPRAY heading thus it avoids the fuel heating up and large release when it succeeds.

The third contributor is PDS Drain down (small leak) (Less than 0.1 percent of the total LRF). Input for this PDS is SFP leak due to hydrogen burning (reactor challenge). FLSS/FLSR is not credited since FLSS piping is damaged due to hydrogen burning.

Figure 25.9.7-2 shows contribution to LRF by Level 1 PSA end states. Significant accident sequences, which contribute higher than 1 percent of the LRF, are shown in Table 25.9.7-2. The significant accident sequences correspond to boil-off sequences in SFP Level 2 PSA.

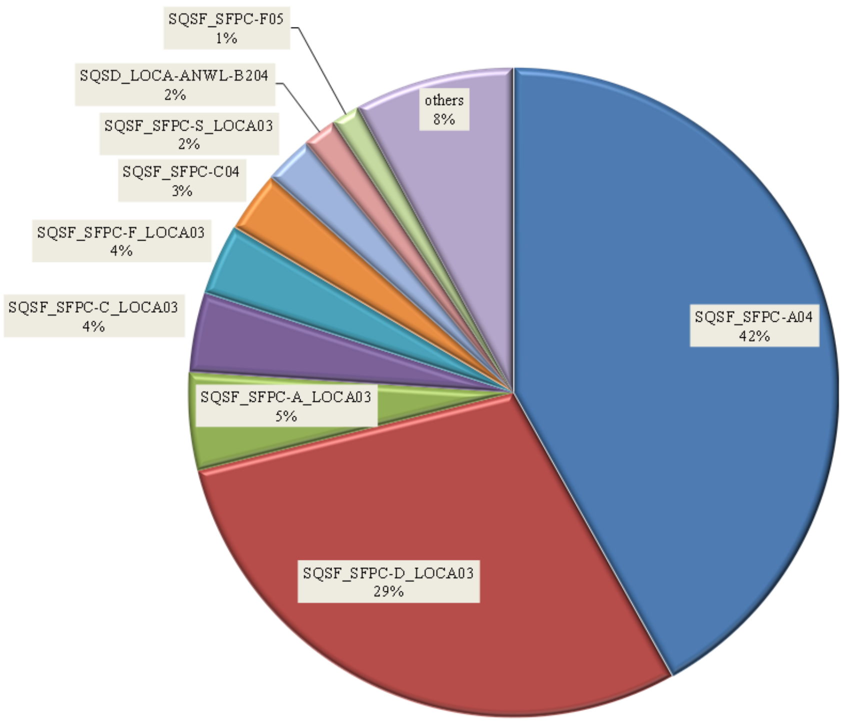

Figure 25.9.7-2 Contribution to LRF by Level 1 PSA End States

Table 25.9.7-2 The Significant Accident Sequences

#	Sequence ID	POS	Contribution	Treatment in SFP Level2 PSA
1	SFPC-A04	A	41.9 %	Boil off
2	SFPC-D_LOCA03	D	29.3 %	Boil off
3	SFPC-A_LOCA03	A	4.9 %	Boil off
4	SFPC-C_LOCA03	C	4.0 %	Boil off
5	SFPC-F_LOCA03	F	3.5 %	Boil off
6	SFPC-C04	C	3.0 %	Boil off
7	SFPC-S_LOCA03	S	2.4 %	Boil off
8	LOCA-ANWL-B204	B2(Reactor)	1.7 %	Boil off
9	SFPC-F05	F	1.4 %	Boil off
	Total	-	92.1 %	-

(c) Significant Accident Progression Sequences/Release category

A total of 7 quantified accident progression sequences exceed the truncation value of 1E-14 /y.

Figure 25.9.7-3 illustrates the LRF contribution for all accident sequences in the form of a bar chart.

The all accident sequences which contribute to LRF are described below.

Rank 1: <u>BO-SFP-02</u>

BO-SFP-02 is an accident progression sequence in PDS Boil-off.

- Fuel uncovery occurs due to boil-off.
- SFP spray after fuel uncovery by FLSS fails.
- Blowout panel is not opened
- Hydrogen burning does not occur but a significant amount of fission products are released through the leak path of secondary containment with steam.

This accident progression sequence has 53 percent of total LRF. This sequence has 53 percent contribution of PDS Boil-off.

Rank 2: <u>BO-SFP-LOCA02</u>

BO-SFP-LOCA02 is an accident progression sequence in PDS Boil-off.

- Fuel uncovery occurs due to boil-off. This boil-off scenario is induced by reactor side BOC/ISLOCA.
- SFP spray after fuel uncovery by FLSS fails.
- Blowout panel is successfully opened by R/B pressurisation due to BOC/ISLOCA.
- Hydrogen burning does not occur but a significant amount of fission products are released through the leak path of secondary containment with steam.

This accident progression sequence has 46 percent of total LRF. This sequence has 46 percent contribution of PDS Boil-off.

Rank 3: <u>BO-SFP03</u>

BO-SFP03 is an accident progression sequence in PDS Boil-off.

- Fuel uncovery occurs due to boil-off.
- SFP spray after fuel uncovery by FLSS fails.
- Blowout panel is not opened.
- Hydrogen burning occurs; therefore, a significant amount of fission products are released through the damaged location of secondary containment.

This accident progression sequence has 0.5 percent of total LRF. This sequence has 0.5 percent contribution of PDS Boil-off.

Rank 4: <u>BO-SFP-LOCA03</u>

BO-SFP-LOCA03 is an accident progression sequence in PDS Boil-off.

- Fuel uncovery occurs due to boil-off. This boil-off scenario is induced by reactor side BOC/ISLOCA.
- SFP spray after fuel uncovery by FLSS fails.
- Blowout panel is successfully opened by R/B pressurisation due to BOC/ISLOCA.
- Hydrogen burning occurs in spite of opened blowout panel so a significant amount of fission products are released through the damaged location of secondary containment.

This accident progression sequence has 0.4 percent of total LRF. This sequence has 0.4 percent contribution of PDS Boil-off.

Rank 5: <u>DDL-SFP02</u>

DDL-SFP02 is an accident progression sequence in PDS Drain down (large leak).

- Fuel uncovery occurs due to SFP large leak.
- SFP spray after fuel uncovery by FLSS fails.
- Blowout panel is not opened.
- Hydrogen burning does not occur, but a significant amount of fission products are released through the leak path of secondary containment with steam.

This accident sequence has 0.2 percent of total LRF. In addition, this is an accident sequence which constitutes LERF. This sequence has 99.5 percent contribution of PDS Drain down (large leak).

Rank 6: <u>DDS-SFP05</u>

DDS-SFP05 is an accident progression sequence in PDS Drain down (Small leak).

- Fuel uncovery occurs due to SFP large leak.

- SFP spray after fuel uncovery by FLSS fails.
- Hydrogen burning occurs; therefore, a significant amount of fission products are released through the damaged location of secondary containment.

This accident sequence has 0.02 percent of total LRF. In addition, this is an accident sequence which constitutes LERF. This sequence has 100 percent contribution of PDS Drain down (large leak).

Rank 7: DDL-SFP03

DDL-SFP03 is an accident progression sequence in PDS Drain down (Large leak).

- Fuel uncovery occurs due to SFP large leak.
- SFP spray after fuel uncovery by FLSS fails.
- Blowout panel is not opened
- Hydrogen burning occurs; therefore, a significant amount of fission products are released through the damaged location of secondary containment.

This accident sequence has less than 0.01 percent of total LRF. In addition, this is an accident sequence which constitutes LERF. This sequence has 0.5 percent contribution of PDS Drain down (large leak).

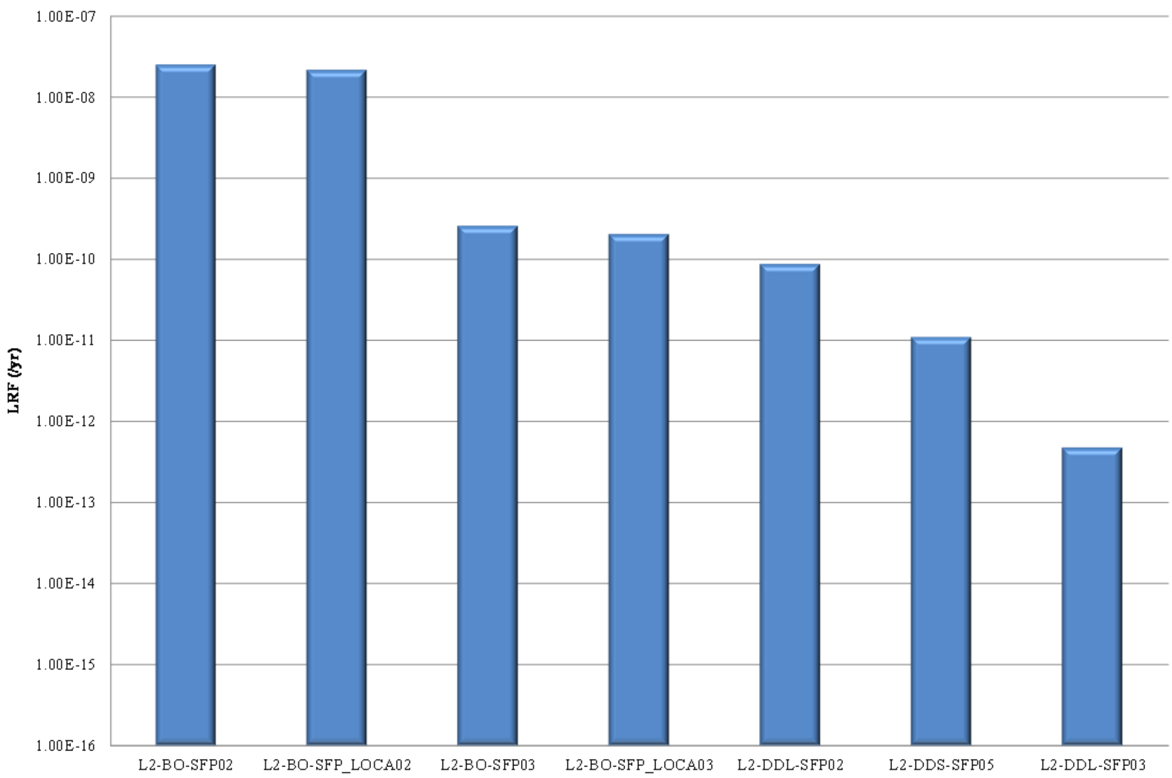

Figure 25.9.7-3 LRF by all Accident Progression Sequence

(d) Significant basic events

Significant basic event: a basic event that contributes significantly to the computed risks for a specific fault group. For internal events, this includes any basic event that has an F-V importance based on LRF greater than 0.005.

(1) Component important to safety (High F-V importance)
FLSR initiation
The highest F-V event is FLSR-SD "FLSR (Mobile Injection Facility) Unavailability": The F-V is 8.08E-01. FLSR can cope with all accident sequences except for SFP Large leak due to SFP catastrophic failure.

FLSS spray header
The second F-V event is E71-SR_-PG-__-D009 "Strainer D009 Plugged": The F-V is 1.21E-01. This basic event denotes the plug of FLSS spray header by using strainer failure rate. This is due to the FLSS effectiveness on each accident sequence. FLSS can cope with all accident progression sequence. (When hydrogen combustion occurs, FLSS is not credited.)

FLSR initiation
The third highest F-V event is FLSR-SD_ST "FLSR (Mobile Injection Facility) Unavailability": The F-V is 6.92E-02. The difference between this event and the top F-V event is shown below.

- It is assumed FLSR is required to be in service (connected to the injection point) before starting POS B-2 and until completion of POS C. This enables credit of FLSR in the accident sequences which do not have the time window more than 8 hours for FLSR. This is modelled as "FLSR-SD ST".
- In other POSs, FLSR is credited as "FLSR-SD" for the accident sequences which have the time window for at least 8 hours for FLSR.

Post-Initiator HFEs

Rank	HFE	F-V	Description
1	FLSR-SD	8.08E-01	FLSR (Mobile Injection Facility) Unavailability
2	___-HFE-FC-FL2	6.92E-01	Failure of manual initiation of FLSS
3	___-HFE-FC-FL	6.92E-01	Failure of manual initiation of FLSS
4	___-HFE-MV-FP	2.48E-01	Failure of opening FP valve
5	___-HFE-FC-FP	1.55E-01	Failure of manual initiation of FP
6	FLSR-SD_ST	6.92E-02	FLSR (Mobile Injection Facility) Unavailability (Line-up before IE)
7	___-HFE-CB-SE	4.48E-02	Circuit breaker selection error

Manual initiation of FLSS, FLSR and Fire Protection system have high importance because redundancy of Class 1 system is degraded. Except for FLSS and FLSR, the FV importance of post-initiator HFEs still remain about 0.4, such as manual valve operation or initiation of Fire Protection system.

FLSR failure probability is represented by the associated human error probabilities (FLSR-SD, FLSR-SD_ST). It is assumed FLSR is required to be in service (connected to the injection point) before starting POS B-2 and until completion of POS C [Ref-25.75]. This enables credit of FLSR in the accident sequences which do not have the time window more than 8 hours for FLSR. In other POSs, FLSR is credited for the accident sequences which have the time window for at least 8 hours for FLSR [Ref-25.75]. These HFEs have high importance values because they are independent from Class 1 support systems.

Pre-Initiator HFEs

Rank	HFE	FV	Description
1	E71-HFE-CL-__-MVF032	1.86E-02	Manual valve F032 left close
2	E71-HFE-CL-__-MVF035	1.86E-02	Manual valve F035 left close

These events disable FLSS (No. 1, 2) however which have small F-V importance.
Manual valves left closed E71-F032 or F035 are related to FLSS injection.

Initiating Event HFEs (TYPE B)

Rank	HFE	F-V	Description
1	___-HFE-SB-AC_A	4.07E-01	Loss of a Class 1 AC bus
2	ANWL-M_B2	1.72E-02	LOCA (mechanical) above normal water level

The human error "___-HFE-SB-AC_A" causing initiating events have high FV. The reason is that this initiator has wider impacts on the mitigation systems designed as Class 1.

(e) Contribution from reactor challenge

Rank	ID	F-V	Description
1	FACTOR_RBS_OK	3.60E-02	Reactor challenge (R/B Steaming)
2	FACTOR_RBS_SD_RB	6.56E-03	Reactor challenge (R/B Steaming, During Shutdown)
3	FACTOR_SD_DW	1.72E-03	Reactor challenge (Strain and Deformation)
4	FACTOR_RBS_SD_OP	1.54E-03	Reactor challenge (R/B Steaming, During Shutdown)

The highest F-V reactor challenge sequence is R/B steaming after OK PCV* failure in the IEAP Level 2 PSA. In the SFP Level 2 PSA, R/B steaming losses function of R/B equipment.

The second highest F-V reactor challenge sequence is R/B steaming during shutdown. In this sequence, the release path of steam and fission products is from the operating deck floor.

The third highest F-V reactor challenge sequence is strain and deformation after containment leakage from D/W. When strain and deformation occurs during at power events, SFP PSA treats it as loss of inventory.

*OK PCV failure: These accident sequences have successful reactivity control, successful water injection to reactor and failure of decay heat removal in the containment. The reactor is cooled by equipment located outside R/B such as Feedwater system, Condensate water system, FLSS or FLSR because containment is failed by overpressure and this containment failure causes adverse effect on the SSCs in R/B.

25.9.7.2 Frequency of Each Release Category

The quantification of frequency in CET and source term analysis are summarised in Table 25.9.7-3. The Sum of frequencies of each release category is used for Level 3 PSA. The results of Level 2 PSA consist of radioactive release frequencies and source term.

Total LRF for the UK ABWR SFP is 4.78E-08 /y.

Large Early Release Frequency

LERF is calculated as the sum of the following release frequencies.

Release category for PDS: drain down (large leak)

DDL-SFP02: In this release category, spray has failed and the blowout panel opening has also failed. However, hydrogen burning does not occur. This release category is categorised into large release in early phase.

DDL-SFP03: In this release category, spray has failed and the blowout panel opening has also failed. Finally, hydrogen burning occurs. This release category is categorised into large release in early phase.

Total LERF for the UK ABWR SFP is 8.85E-11 /y.

Table 25.9.7-3 Radioactive Release Frequencies and Source Term Information

No.	Release Category	Frequency (/y)	Source Term
1	BO-SFP02	2.54E-08	Table 25.9.6-2 Case 1 Reactor side RC should be combined. Frequency of reactor challenge is 1.0E-9 /y *1
2	BO-SFP_LOCA02	2.19E-08	Table 25.9.6-2 Case 1 Reactor side RC should be combined. Frequency of reactor challenge is 5.1E-11 /y *3
3	BO-SFP03	2.61E-10	Table 25.9.6-2 Case 1 Reactor side RC should be combined. Frequency of reactor challenge is 9.7E-12 /y *1
4	BO-SFP_LOCA03	2.06E-10	Table 25.9.6-2 Case 1 Reactor side RC should be combined. Frequency of reactor challenge is 4.4E-14 /y *3
5	DDL-SFP02	8.80E-11	Table 25.9.6-2 Case 3 Reactor side RC should be combined. Frequency of reactor challenge is 9.5E-11 /y *2
6	DDS-SFP05	1.10E-11	Table 25.9.6-2 Case 3 Reactor side RC should be combined. Frequency of reactor challenge is 1.1E-11 /y *2
7	DDL-SFP03	4.78E-13	Table 25.9.6-2 Case 3 Reactor side RC should be combined. Frequency of reactor challenge is 4.1E-13 /y *2
8	Total LRF	4.78E-08	-
9	Total LERF	8.85E-11	-

*1: LRF caused by PDS VIII sequence in the reactor shutdown PSA is summarised.

*2: All cutsets are caused by reactor challenge.

*3: This frequency is derived from importance of the sequence tag of non-success sequences in the reactor PSA.

25.9.8 Analysis for Level 3 PSA

25.9.8.1 Radionuclide release categories

Ten RCs were defined in the SFP IE Level 2 PSA. These were derived from simple CETs developed from considerations of availability of SFP spray system, opening of the passive blow-out panel between the Reactor Building and the environment, and the potential for a hydrogen burn. The ten RCs were allocated to three Level 3 cases designated F1 to F3. The Level 3 cases, RCs and their frequencies of occurrence, and the representative severe accident sequence are presented in Table 25.9.8-1.

Table 25.9.8-1 Release Categories, Level 3 PSA Cases. Representative SAA Sequences and Frequency of Occurrence for SFP IE

Level 3 PSA Main Case	Level 2 PSA release category	Representative SAA sequence	Level 3 PSA sub-cases and frequencies (/y)			Total Frequency (/y)
			SFP case alone	SFP + Bypass (P13)	SFP + reactor challenge	
F1	BO-SFP02	SFP Boil Off (due to failure of injection and spray systems)	2.44E-08	-	1.00E-09	2.54E-08
	BO-SFP-LOCA02		2.18E-08	5.10E-11	-	2.19E-08
	BO-SFP03		2.51E-10	-	9.70E-12	2.61E-10
	BO-SFP-LOCA03		2.06E-10	4.40E-14	-	2.06E-10
F2	DDS-SFP05	SFP Drain Down (due to LOCA + due to failure of injection and spray systems)	-	-	1.10E-11	1.10E-11
F3	DDL-SFP02	Catastrophic Failure	-	-	8.80E-11	8.80E-11
	DDL-SFP03		7.00E-14	-	4.08E-13	4.78E-13

As shown in Table 25.9.8-1, for some RCs, a percentage of the frequency is combined with a reactor challenge event, i.e. these RCs are combined with some RCs from IE at Power. In some cases, the IE at Power RC is uniquely defined, (e.g. SFP IE release category BO-SFP-LOCA02 is combined with IE at Power RC P14, "Containment Bypass"). In other cases the reactor challenge may represent several IE at Power RCs.

The representative SAA sequences are assumed to result in fuel melt (with the exception of a sequence in the Level 2 PSA for catastrophic failure condition which assumes SFP spray success and no fuel melt) and to result in large releases to the environmental. In the SAA, the SGTS is assumed to fail and the blow out panel open, providing a direct path to the environment.

(1) Mass fraction releases and release profiles

The SAA for SFP IE was performed using MAAP 5, which allows the behaviour of the hot fuel bundles and cold fuel bundles (i.e. recently unloaded fuel and fuel that has been cooled for at least one operational cycle) to be modelled. However, due to the differences in inventories and the thermal conditions seen in the hot and cold fuel bundles, the combined mass release predictions generated by the MAAP5 code are more complex to interpret – rendering the matrix approach used for the IE at Power and IE at Shutdown cases inapplicable. To determine releases to the environment, an approach is used which takes account of the mass of fuel melted in both the hot and cold bundles and their individual inventories, and then applies an Effective Release Fraction (ERF) to the summated activity.

(2) Releases of radioactivity to the environment (source term)

For consistency with the IE at Power and IE at Shutdown, the MAAP5 fission product groups were collapsed into the same groups used in MAAP4.07. However, unlike in the case of IE at Power, some simplifying assumptions were made in the Level 3 PSA for SFP IE. One of the key simplifications is the assumption that the fission products are released in a single phase of one hour. The single phase release approach minimises radioactive decay and previous sensitivity cases have indicated that it generally leads to higher consequences.

The source term input for SFP IE is estimated using the following approach:

(i) Effective Release Fractions (ERF), i.e. the fraction of mass lost from the fuel that is released to environment, are calculated for each Source Term Group (STG). This is then applied separately to the mass lost from hot and cold fuel bundles. The retention of an aerosol by the reactor building is assumed to depend only on its STG, and not whether it originated in the hot or cold fuel bundles.

(ii) The release estimate (in Bq) for each radionuclide is then calculated using:

  (a) The mass release fractions estimated for each fission product group, for the hot and cold fuel bundles separately, as described above

  (b) The separate, representative radionuclide inventories for the hot and cold fuel bundles

(iii) The total release estimate (in Bq) for each radionuclide is then calculated as the sum of the hot bundles and cold bundles release estimates. This total is input into PC COSYMA in a single release phase.

For ‘Combined Events’, where releases arise from both SFP IE and IE at Power as part of a more widespread plant event, the releases from the component events are summed in the Level 3 PSA. For simplicity, the releases from IE at Power contributors are represented by release category P14 which is one of the highest IE at Power sequences. Thus the ‘SFP alone’ and ‘SFP + P14’ cases bracket the range of likely releases. Again a single release phase is considered.

25.9.9 Results (SFP Level 3)

25.9.9.1 Facility dose bands

The dose allocations for each Level 3 PSA case for SFP IE are presented in Table 25.9.9-1. The facility dose band allocations, which are determined based on the highest summated long term dose (cloud gamma, inhalation and long term ground gamma) for each case.

Table 25.9.9-1 Allocation to Facility Dose Bands for SFP Internal Events

Level 3 PSA case identifier	Dose Band
F1	> 1 Sv
F2	> 1 Sv
F3	> 1 Sv
F1 + RC Challenge	> 1 Sv
F1 + P13	> 1 Sv
F2 + RC Challenge	> 1 Sv
F3 + RC Challenge	> 1 Sv

Table 25.9.9-2 presents the assessment against facility dose bands for the SFP IE. All the Level 3 PSA cases are allocated to the > 1 Sv dose band and the summated frequency contribution is 4.79E-08 /y or 4.8 percent of the BSO.

It can be seen that two release categories contribute almost all of the frequency in the highest facility dose band:

- Release category BO-SFP-02 is the largest contributor at 2.44E-08 /y or 51.0 percent of the total,
- Release category BO-SFP-LOCA02 contributes 2.18E-08 /y or 45.6 percent of the total.
- Both release categories are represented by Level 3 PSA case F1.
- Release category BO-SFP-02 in combination with a reactor challenge contributes 1.00E-09 /y or 2.1 percent of the total.

This combination is represented by the Level 3 PSA case F1 + P13.

Table 25.9.9-2 Assessment Against Facility Dose Bands for the SFP IE Release Categories

Effective dose (Sv)	Release categories	Frequency (/y)	Percentage of BSO	BSO	BSL
0.0001 – 0.001	-	-	-	1.0E-2	1
0.001 – 0.01	-	-	-	1.0E-3	1.0E-1
0.01 – 0.1	-	-	-	1.0E-4	1.0E-2
0.1 – 1	-	-	-	1.0E-5	1.0E-3
> 1	BO-SFP02 (+RC) BO-SFP03 (+RC) BO-SFP-LOCA02 (+RC) BO-SFP-LOCA03 (+RC) DDS-SFP05 (+RC) DDL-SFP02 (+RC) DDL-SFP03 (+RC)	4.79E-08	4.79 %	1.0E-6	1.0E-4
Summated frequency of food bans /y		4.79E-08			

25.9.9.2 Individual risk

Conditional individual risks are calculated for each release category for SFP IE, at distances of 400 m, 1,000 m and 1,500 m from the site to illustrate how the conditional individual risk varies with distance over the range of likely off site locations with continuous occupancy.

The contribution of each release category to the overall risk from SFP IE is given in Table 25.9.9-3.

The summated individual risk at 1 km from SFP IE is 5.63E-09 /y or 0.6 percent of the BSO. It is noted that all these individual risk values have been calculated assuming a single release phase. This assumption is likely to result in lower individual risk values than if multiple phases were able to be used in the conditional risk calculations. It is currently thought that use of multiple phases would increase the conditional individual risk close to the site by a factor of about 2. However, as the frequency of all release categories is low (at 4.79E-08 /y), even the worst-case assumption would only be a small fraction of the BSO.

Figure 25.9.9-1 shows the contributions or the releases categories to the individual risk. It can be seen that three release categories contribute almost all of the individual risk:

- Release category BO-SFP-02 is the largest contributor at 2.87E-09 /y or 50.9 percent of the total,
- Release category BO-SFP-LOCA02 contributes 2.57E-09 /y or 45.6 percent of the total,
- Release category BO-SFP-02 in combination with Reactor Challenge contributes an additional 1.23E-10 /y or 2.2 percent of the total.

Table 25.9.9-3 Individual Risk of Fatal Health Effects Near to the Site for the SFP IE with Minimal off Site Protective Actions

Release Category	Individual risk of fatal health effects (/y)			Contribution to Total Risk at 1 km
	400 m	1,000 m	1,500 km	
BO-SFP02	3.86E-09	2.87E-09	2.49E-09	50.9 %
BO-SFP03	3.97E-11	2.95E-11	2.56E-11	0.52 %
BO-SFP-LOCA02	3.45E-09	2.57E-09	2.23E-09	45.6 %
BO-SFP-LOCA03	3.25E-11	2.42E-11	2.10E-11	0.43 %
BO-SFP02 + RC	1.65E-10	1.23E-10	1.08E-10	2.18 %
BO-SFP03 + RC	1.60E-12	1.19E-12	1.05E-12	0.02 %
BO-SFP-LOCA02 + RC	8.42E-12	6.26E-12	5.51E-12	0.11 %
BO-SFP-LOCA03 + RC	7.26E-15	5.40E-15	4.75E-15	<0.01 %
DDS-SFP05 + RC	1.83E-12	1.36E-12	1.19E-12	0.02 %
DDL-SFP03	7.91E-15	4.78E-15	3.40E-15	<0.01 %
DDL-SFP02 + RC	1.30E-11	9.25E-12	7.86E-12	0.16 %
DDL-SFP03 + RC	6.04E-14	4.29E-14	3.64E-14	<0.01 %
Total individual risk: /y	**7.57E-09**	**5.63E-09**	**4.89E-09**	
Total as % of BSO	**0.75 %**	**0.56 %**	**0.49 %**	
BSO	1.00E-06			
BSL	1.00E-04			

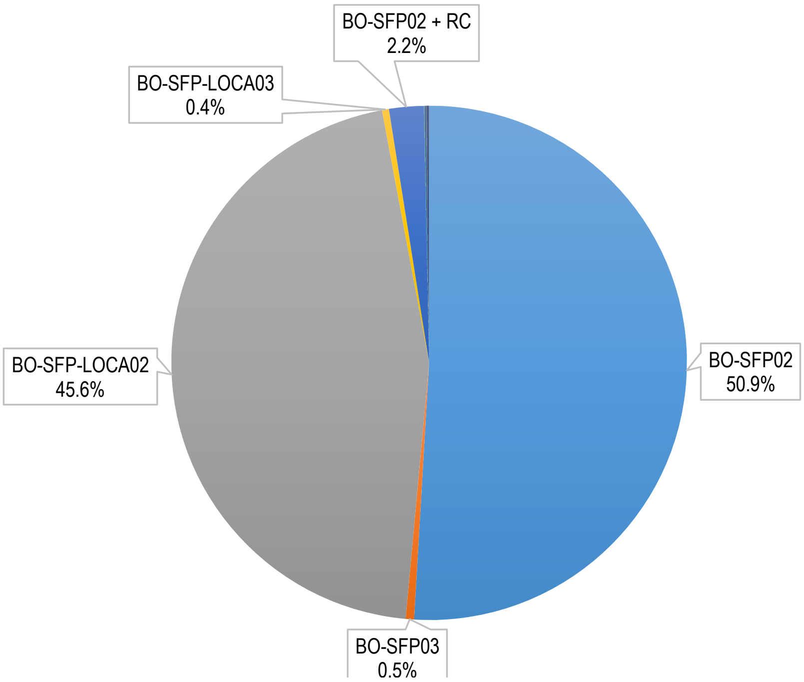

Figure 25.9.9-1 Contribution of Release Categories to the Individual Risk at 1 km for SFP IE Leading to Fuel Melt

25.9.9.2 Societal risk

The summated mean numbers of health effects predicted in the UK population for SFP IE are calculated. The breakdown between the mean number of early fatal health effects and the mean number of notional late fatalities is given. The conditional probability of exceeding 100 notional late fatalities is also given and it can be seen that these are close to unity in all the Level 3 PSA cases.

Table 25.9.9-4 presents the conditional societal risk for SFP IE Level 3 PSA cases. All cases exceed the Target 9 threshold by a substantial margin. A large number of early fatal health effects are predicted in these calculations with minimal offsite protective actions.

The total frequency of exceeding the Target 9 threshold is 4.79E-08 /y or 48 percent of the BSO. Figure 25.9.9-2 shows that three release categories contribute almost all of the frequency above the societal risk threshold:

- Release category BO-SFP-02 is the largest contributor at 2.44E-08 /y or 51.0 percent of the total,
- Release category BO-SFP-LOCA02 contributes 2.18E-08 /y or 45.6 percent of the total,
- Release category BO-SFP-02 in combination with Reactor Challenge contributes an additional 1.00E-09 /y or 2.1 percent of the total.

Table 25.9.9-4 Societal Risk for the SFP IE with Minimal off Site Protective Actions

Release Category	Release category frequency (/y)	Frequency above Target 9 threshold (/y)	Contribution to total frequency
BO-SFP02	2.44E-08	2.44E-08	51.0 %
BO-SFP03	2.51E-10	2.51E-10	0.53 %
BO-SFP-LOCA02	2.18E-08	2.18E-08	45.6 %
BO-SFP-LOCA03	2.06E-10	2.06E-10	0.43 %
BO-SFP02 + RC	1.00E-09	1.00E-09	2.09 %
BO-SFP03 + RC	9.70E-12	9.70E-12	0.02 %
BO-SFP-LOCA02 + RC	5.10E-11	5.10E-11	0.11 %
BO-SFP-LOCA03 + RC	4.40E-14	4.40E-14	<0.01 %
DDS-SFP05 + RC	1.10E-11	1.10E-11	0.02 %
DDL-SFP03	7.00E-14	7.00E-14	<0.01 %
DDL-SFP02 + RC	8.80E-11	8.80E-11	0.18 %
DDL-SFP03 + RC	4.08E-13	4.08E-13	<0.01 %
	Total Frequency: /y	**4.79E-08**	
	Total as % of BSO	**47.9 %**	
	BSO	1.00E-07	
	BSL	1.00E-05	

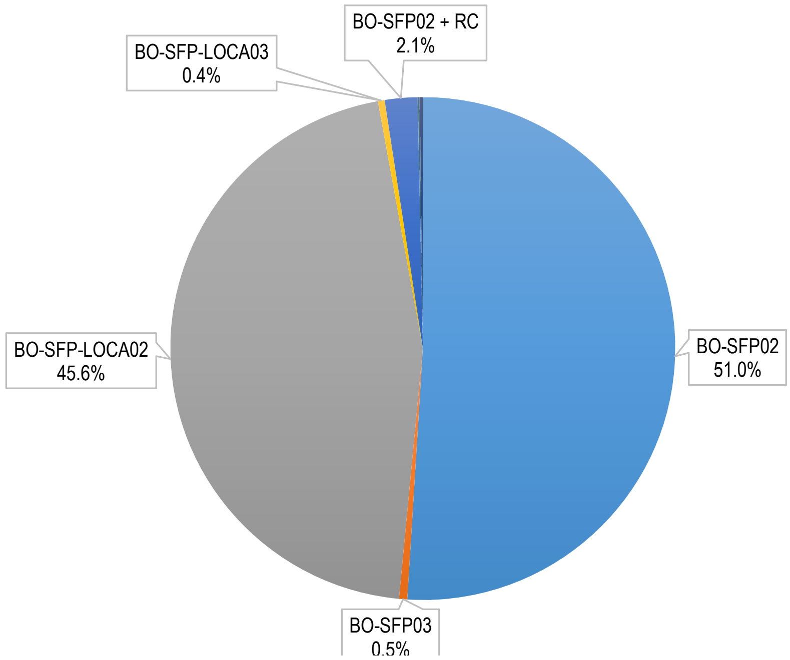

Figure 25.9.9-2 Contribution of Release Categories to the Frequency of Exceeding the Societal Risk Criterion for SFP IE Leading to Fuel Melt

25.9.10 Uncertainty and Sensitivity Analysis

25.9.10.1 Sensitivity Analysis Results (Individual Sensitivity Cases)

Sensitivity Analyses have been performed for each key safety function using quantitative and/or qualitative methods (see Section 25.7.5.2 (b)).

Table 25.9.10-1 shows the summary of the sensitivity analyses.

Table 25.9.10-1 Sensitivity Analysis Cases and Results

No.	Sensitivity Analysis Case	Difference FDF	Difference LRF	Remarks
1	Credit of FLSS spray in Level 1 PSA	-89 %	ε^{*1}	Minimum impact on LRF while significant impact on FDF. Note: LRF is used for the comparison of NSEDP/SAP targets.
2	Effectiveness of FLSS spray	ε^{*1}	ε^{*1}	Minimum impact on the PSA results (PSA model is not changed. This is the discussion using the cutsets in the base case.)
3	Recovery of FPC	-51 %	-43 %	Significant impact on the PSA results. The margin is increased compared with NSEDP/SAP targets.
4	Increase of conditional probability due to reactor challenge	ε^{*1}	ε^{*1}	Minimum impact on the PSA results.
	Decrease of conditional probability due to reactor challenge	ε^{*1}	ε^{*1}	Minimum impact on the PSA results.
5	Effectiveness of SGTS as a countermeasure of FP release	-	-	Top 20 cutsets in the base case are reviewed whether SGTS is credited in the cutset or not. SGTS is credited in only 13th cutset whose contribution is 0.4 percent against top 20 cutsets. It was found that SGTS would not be available in most situations, considering the dependencies such as AC power or the environmental impact of steam.
6	Dependency of Multiple Human Failure Events (HFEs) (Lower Dependency)	-	-38 %	Significant impact on the PSA results. The margin is increased compared with NSEDP/SAP targets.
	Dependency of Multiple Human Failure Events (HFEs) (Higher Dependency)	-	70 %	Significant impact on the PSA results.
7	HRA Report Rev. E	434 %	247 %	Significant impact on the PSA results.

*1: ε = extremely small

25.9.10.2 Uncertainty Analysis

Uncertainty analyses were performed to support robustness of the PSA and design of UK ABWR.

Uncertainty analysis is performed using the Monte Carlo sampling method which generates a probability density function and a cumulative probability function for Fuel Damage Frequency (FDF), Large Release Frequency (LRF) and Large Early Release Frequency (LERF). LERF is evaluated as the sum of the release frequencies in early phase (DDL-SFP02 and DDL-SFP03).

A sample size of 100,000 was used to generate the associated results from the cutsets with the truncation value of 1.0E-14 /y though FDF and LRF converge [Ref-25.90] [Ref-25.76]. The mean values generated based on the sample size are FDF of 4.54E-07 /y, LRF of 5.41E-08 /y and LERF of 1.22E-10 /y as shown in Tables 25.9.10-2, 25.9.10-4 and 25.9.10-6. The results of the internal events at Power PSA show that the mean value of LRF is below the target 7 and 8 BSO of 1E-6 /y. Furthermore, the uncertainty analysis results for each POS are shown in Tables 25.9.10-3 and 25.9.10-5.

In addition, uncertainty analysis for each fission product release category is performed as shown in Table 25.9.10-6.

Table 25.9.10-2 Uncertainty Analysis for Total SFP FDF

Case	Mean	5 %	Median	95 %
FDF	4.54E-07	2.43E-08	1.90E-07	1.63E-06

Table 25.9.10-3 Uncertainty Analysis for FDF by POS

Case	Mean	5 %	Median	95 %
FDF (POS S)	1.19E-08	8.51E-11	2.28E-09	4.85E-08
FDF (POS A)	2.46E-07	9.85E-09	9.52E-08	9.06E-07
FDF (POS C)	3.05E-08	4.78E-10	7.99E-09	1.21E-07
FDF (POS D)	9.38E-08	8.40E-10	2.01E-08	3.72E-07
FDF (POS E)	5.17E-09	4.95E-11	9.30E-10	1.94E-08
FDF (POS F)	6.63E-08	1.81E-09	1.62E-08	2.19E-07

Table 25.9.10 -4 Uncertainty Analysis for Total LRF

Case	Mean	5 %	Median	95 %
LRF	5.41E-08	4.40E-09	2.25E-08	1.79E-07

Table 25.9.10-5 Uncertainty Analysis for LRF by POS

Case	Mean	5 %	Median	95 %
LRF (POS S)	1.25E-09	1.21E-11	2.53E-10	4.84E-09
LRF (POS A)	2.37E-08	1.57E-09	9.27E-09	8.55E-08
LRF (POS B-1)	2.43E-10	3.12E-12	4.22E-11	8.33E-10
LRF (POS B-2)	8.60E-10	1.64E-11	2.11E-10	3.33E-09
LRF (POS C)	3.73E-09	9.72E-11	1.19E-09	1.40E-08
LRF (POS D)	1.50E-08	1.94E-10	3.67E-09	5.97E-08
LRF (POS E)	5.61E-10	9.26E-12	1.31E-10	2.07E-09
LRF (POS F)	5.68E-09	2.18E-10	1.50E-09	1.92E-08

Table 25.9.10-6 Uncertainty Analysis for Release Category

Release Category	Mean	5 %	Median	95 %
BO-SFP02	2.71E-08	2.29E-09	1.17E-08	9.30E-08
BO-SFP03	2.68E-10	2.50E-12	4.48E-11	9.33E-10
BO-SFP_LOCA02	2.35E-08	7.74E-10	7.38E-09	8.86E-08
BO-SFP_LOCA03	2.04E-10	8.89E-13	2.41E-11	7.37E-10
DDS-SFP05	1.46E-11	3.87E-14	1.23E-12	3.75E-11
DDL-SFP02	1.20E-10	5.49E-13	1.62E-11	4.33E-10
DDL-SFP03	3.82E-13	4.87E-17	8.68E-15	9.52E-13

25.9.11 Insights of Sensitivity Analysis and Uncertainty Analysis

The most significant insights from the results of sensitivity and uncertainty analyses are shown below based on the internal events at power results.

Overall, it was found that risk impacts by most of the sensitivity cases are not significantly increased except in three cases as described below.

Sensitivity cases which are increased from the base case by roughly 20 percent or more are identified as shown in Table 25.9.11-1. These cases are considered in further assessments as described below.

Table 25.9.11-1 Sensitivity Results increased from the Base Case by 20 % or more

No.	Sensitivity Analysis Case	Difference		Remarks
		FDF	LRF	
1	Credit of FLSS spray in Level 1 PSA	-89 %	ε*1	Minimum impact on LRF while significant impact on FDF. Note: LRF is used for the comparison with NSEDP/SAP targets.
3	Recovery of FPC	-51 %	-43 %	Significant impact on the PSA results. The margin is increased compared with NSEDP/SAP targets.
6	Dependency of Multiple Human Failure Events (HFEs) (Lower Dependency)	-	-38 %	Significant impact on the PSA results. The margin is increased compared with NSEDP/SAP targets.
	Dependency of Multiple Human Failure Events (HFEs) (Higher Dependency)	-	70 %	Significant impact on the PSA results.

*1: ε = extremely small

- Sensitivity case No.1 identified that the credit of FLSS spray in Level 1 PSA had a large impact on SFP FDF. However, the result indicates that there is no impact on the comparison with the NSEDP/SAP targets because SFP LRF is not changed.
- Sensitivity case No.3 identified that the FPC recovery assumption had a large impact on uncertainty in SFP FDF and LRF. The main contributors to the uncertainty of FPC failure were human error and AC bus failure. This was informed to the designers and risk reduction measures were discussed [Ref-25.11].
- Sensitivity case No.6 identified that HEP dependency assumption contributed significantly to the uncertainty in the SFP PSA. The adequacy of the HEP dependency modelling is reviewed considering the result of this sensitivity analysis.
- The mean values generated by uncertainty analysis are SFP FDF of 4.54E-07 /y and LRF of 5.41E-08 /y. The results of the internal events at Power PSA show that the mean value of LRF is below the target 7 and 8 BSO of 1E-6 /y.

- Six sensitivity analyses have been performed individually. Table 25.9.11-1 shows the sensitivity cases where the impact of SFP FDF and/or SFP LRF is significant.

It is understood that the sensitivity analysis is not used as a compensation for a lack of modelling detail in the base case. For some issues with large uncertainty, it may be difficult to select the best estimate model, or there may be insufficient design details at this point to model with certainty. Sensitivity analyses help to overcome this problem by identifying the issues which need to be dealt with carefully. This demonstrates that PSA is an important tool for design and operation. The PSA and risk insights are used to support the ALARP assessment and are shared with the appropriate design teams in accordance with Hitachi-GE's design process.

25.9.12 Insights from assessment

Level 1 PSA

- POS A has relatively high risk (about 58 percent of the total SFP FDF) because Class 1 division 2 is out of service. Division 1 and 3 are available but division 3 effectiveness is limited for SFP critical safety function.
- POS D has relatively high risk (about 23 percent of the total SFP FDF) because one division of the backup building systems is out of service.
- Due to the partially degraded redundancy of the Class 1 systems, FLSS is important in all POS. The post-initiator HFE that has the highest F-V and RAW is the manual initiation of FLSS.
- FLSR provides considerable risk reduction. It is assumed FLSR is required to be connected to the injection point before an IE during POS C per the recommendation by the experienced system engineer and maintenance planner in the Design Review meeting, so that the redundancy of mitigation systems is improved under the condition Class 1 Divisions 1 and 3 systems are in maintenance. This is consistent with shutdown PSA modelling.
- HFEs (Type B) on Loss of Class 1 AC initiating event have a significant risk contribution in POS A. Review of this human failure event and refinement of this analysis may be effective to develop a more realistic model when the information of this action is available.
- Injection from the Fire Protection system is an important countermeasure when LOOP occurs with loss of Class 1 AC power as the Fire Protection system has a diesel driven pump that can be used to inject water without any AC power. It can be recovered with a relatively long time margin.

Level 2 PSA

- POS A has the highest FDF (about 49 percent of the total LRF). The initiating event "Loss of Class 1 AC due to human error" in POS A contributes to about 41 percent of the total LRF.
- POS D has the second highest LRF (about 31 percent of the total LRF). During this POS only 1 FLSS is available due to the maintenance schedule.
- The third contributor is POS C (about 8 percent of the total LRF). In POS C, only Division 2 is available. Divisions 1 and 3 are out of service and the Division 3 effectiveness is limited. Especially for the POS C, the number of credited systems is limited subsequently FLSS importance is relatively high from the view point of the SFP Level 1 PSA accident sequence. In this case, not only human failure events but also mechanical failures have significant contributions. In the Level 2 PSA treatment, SFP spray effect is limited because the FLSS mechanical failure in Level 1 PSA cutset.

- Despite the highest decay heat, POS S has the small LRF due to the most redundant mitigation systems and the shortest duration.
- PDS Boil-off has the highest LRF (about 99.8 percent of the total LRF). Summation of the input for this PDS i.e., FDF in the Level 1 SFP PSA for boil off, FDF in Level 1 shutdown PSA for accident class VIII (eight) and loss of SFP cooling due to R/B steaming (reactor challenge) is 4.6E-07. LRF for this PDS is about 10 percent of the input frequency. This is because a large part of the cutsets for this PDS includes the human failure event for FLSS and it can be credited again in the Level 2 PSA in the SFP_SPRAY heading. Success of SFP SPRAY prevents the fuel heating up and large release.
- The highest F-V reactor challenge sequence is R/B steaming after OK PCV* failure in the IEAP Level 2 PSA. In the SFP Level 2 PSA, R/B steaming causes loss of function of R/B equipment.
- *OK PCV failure: An accident sequence with successful reactivity control, successful water injection to reactor and failure of decay heat removal in the containment. Reactor is cooled by the equipment located outside R/B such as Feedwater system, Condensate water system, FLSS or FLSR because containment is failed by overpressure and this containment failure causes adverse effects on the components in the R/B.
- The second highest F-V reactor challenge sequence is R/B steaming during shutdown. In this sequence, the operating deck floor is the release path of steam and fission products. In the SFP PSA, R/B steaming causes loss of function of R/B equipment.
- The third highest F-V reactor challenge sequence is strain and deformation after containment leakage from the D/W. When strain and deformation occurs during at Power events, the SFP PSA treats it as loss of inventory.
- Simultaneous SFP and Reactor LERF have a small risk contribution.
- It can be seen that the analysis is adequate to identify the significant risk contributors. The implications associated with the PSA and the dominant contributors to risk in terms of following are discussed.
  - Systems that are important to have continued availability
  - Systems whose degradation would lead to increased risk
  - Any design or procedural changes that can be enhanced from the analysis

- Systems that are important to have continued availability/ Systems whose degradation would lead to increased risk: Based on the results of F-V or RAW importance analysis, resulting high importance systems are almost the same as that of Level 1 SFP PSA. This is because the FLSS (spray) is the unique system to decrease the LRF or LERF and the other injection systems are effective to decrease the FDF and eventually decrease the LRF/LERF.
- As mentioned the above, countermeasures to decrease the FDF are also effective to decrease the LRF/LERF. Therefore, design or procedural changes that can be enhanced from the analysis of Level 1 SFP PSA are also applicable to Level 2 SFP PSA.

25.9.13 Key Assumptions and Study Limitations

Assumptions in the SFP PSA were made in the development phase. They relate to each aspects of PSA, such as the identification of initiating events, the definition of success criteria, reliability data selection, containment performance, accident progression and so on. In addition, when the level of plant requirements specification limits the detail of the PSA, then appropriate assumptions were developed to allow completion of the PSA.

Starting from assumptions, sensitivity analyses have been performed. Among assumptions in the SFP PSA, key assumptions, which have comparatively large impact on the result, have been listed from the result of sensitivity analyse.

The key assumptions considered in the SFP PSA are listed below.

- The SFP spray is assumed to be conducted from 0.5 hours after an accident in the accident progression analysis in case 5 (see Table 25.9.6-2).
- Recovery actions such as restoration of RHR or FPC etc. are not credited because it is difficult to identify a failure mode of the failed component.
- For Initiating Events such as reactor fault impacting the SFP, conditional probabilities for the effects on SFP are assumed.
- Dependency of Multiple Human Failure Events are assumed to be zero dependency.

For the Level 3 PSA results, care should also be taken in interpreting the predictions for the combined events, where two SAA sequences with very different accident progression behaviour are simply summed. Due to constraints in manipulating the data, this case is currently represented as a single release phase. It is noted that this aspect of the Level 3 PSA is considered a “first of a kind” analysis in the UK and, consequently, the methodology may be revised in subsequent updates of the Level 3 PSA.

25.10 PSA for Internal Hazards

The PSA for the GDA assessed all types of initiating events including internal events, internal hazards and external hazards. The internal hazard PSA provides an integrated, structured safety analysis to assess the plant risk, identify potential plant vulnerabilities and quantify the risk in support of the ALARP demonstration. The quantitative results of PSA are used to verify compliance with the safety goals or criteria, which are formulated in terms of quantitative estimates of core damage frequency, frequencies of radioactive releases of different types and individual and societal risks.

Section 25.10 describes Level 1, Level 2, and Level 3 PSA results associated with internal hazards for the UK ABWR at the GDA phase. Internal hazards considered in the post GDA phase are outside the scope of the prioritisation analysis and will be deferred until the post GDA phase.

Potential internal hazards were identified and then screening was carried out. The candidate internal hazards then were prioritised to assess their risk. Furthermore, the candidates were divided into GDA and post GDA phases. The methodology and the results of prioritisation are described in Section 25.10.1.

Procedures to assess the internal hazards vary according to the types of internal hazard, thus the assessment was conducted for each hazard candidate. The assessment results are shown in Sections 25.10.2 to 25.10.5.

25.10.1 Basic Approach on PSA for Internal Hazards

In this Section, the approach for performing the prioritisation of internal hazards for detailed analysis in the UK ABWR PSA for GDA is described, including the qualitative and quantitative screening analyses performed in order to identify the list of hazards to be assessed as part of internal hazards PSA.

25.10.1.1 Internal Hazard Definition

Internal hazards are defined as events that originate from sources located on the site of the nuclear power plant, either inside or outside plant buildings. Examples of internal hazards are internal fires, internal floods, turbine missiles, on-site transportation accidents and releases of toxic substances from on-site sources. The origins of internal hazards are man-made and the future licensee has some form of control over the cause of the resulting initiating events. Terrorist or other malicious acts are part of the security assessment and are excluded from this assessment.

Internal hazards can damage plant components and thus generate accident sequences that might lead to core damage (or to other end states). In general, it is more difficult to define the magnitude of a hazard and its consequences than for an internal plant fault, because internal hazards have the potential to affect many different pieces of equipment simultaneously and adversely impact plant personnel.

For the UK ABWR, the PSA for internal hazards is an extension of the design basis hazard analyses, which also addresses all potential internal hazards.

25.10.1.2 Technical Approach and Steps for Prioritisation of Hazards

The objective of the internal hazards PSA is to determine all significant contributing factors to the risks arising from the facility and to evaluate the extent to which the overall design is well balanced and meets probabilistic safety criteria where these have been defined. The assessment of the internal hazards was performed consistent with the expectations of the IAEA GSR Part 4 [Ref-25.92], and IAEA SSG-3 [Ref-25.93]. A successive screening process as recommended in IAEA SSG-3 was followed to minimise the emphasis on internal and external hazards whose contribution to risk is low and thus to focus the analysis on hazards that are risk significant.

A consistent approach was applied to the identification of hazards for the deterministic analysis and the PSA and in the analysis of their contribution to the risks. The main stages of the prioritisation of hazards include (SSG-3 [Ref-25.93]):

(1) Collection of initial information on internal hazards,

(2) Hazard identification, including single and combined hazards,

(3) Hazard screening analysis, both qualitative and quantitative,

(4) Bounding assessment,

(5) Detailed analysis and PSA modelling.

The screening approach defined in SSG-3 describes three processes: The first process is where items can be screened out (removed from the analysis) or screened in (retained in the analysis). For this first step, items are normally screened out qualitatively, where the hazards are not physically possible or realistic, e.g. sea water level increase will not impact an inland facility. The second process is where the hazard is grouped with or bounded by another hazard, i.e. the characteristics are less severe than the bounding hazard. The third process is where a hazard is carried forward to be assessed in more detail as the hazard at the stated frequency is likely to have a significant contribution to Core Damage Frequency (CDF) and those hazards not selected are retained at the frequency at which they are screened.

25.10.1.3 Collection of Initial Information on Internal Hazards

As a starting point of the prioritisation of hazards and thus of the Level 1 PSA for internal hazards, all available information specifically relating to the internal hazards was collected. The following information was used:

- Design information relating to internal hazards as considered in the safety analysis report. This allows identification of defensive measures taken during design of the plant against internal hazards including:
  - resistance / strength of buildings against fires, explosions, blasts, floods etc., protection barriers,
  - fire protection measures: door and wall ratings, fire detection, firefighting, separation,
  - flood doors, sumps (lay-out drawings),
  - water/fire/explosion/LOCA resistant equipment,
  - piping characteristics with respect to pipe whip, leakage (room data sheets or information from plant equipment database).
- Lists and layout of plant buildings, structures, systems and components (lay-out drawings).
- Plant layout and topography of the site and surroundings (plot plan).

- Information on the location of pipelines (trench route map), transportation routes and on-site storage facilities for hazardous materials.

25.10.1.4 Qualitative Screening and Grouping of Hazards

The complete list of hazards to consider was established. In the list 42 hazards are identified, 38 from SE-GD-0193 [Ref-25.91] and other 4 from IAEA SSG-3 [Ref-25.93]. Grouping and screening criteria are defined in order to perform this analysis.

(1) Screening Process for Internal Hazards

The assessment of the internal hazards was performed in a manner consistent with the expectations of the IAEA GSR Part 4 [Ref-25.92], and IAEA SSG-3 [Ref-25.93]. The screening process defined in SSG-3 is shown below;

- The first process is where items can be screened out (removed from the analysis) or screened in (retained in the analysis).
- The second process is where the hazard is grouped with or bounded by another hazard. This step minimises the number of hazards to analyse to focus on the important hazards.
- The third process is where a hazard is carried forward to be assessed in more detail, as the hazard at the stated frequency is likely to make a significant contribution to CDF and the hazards not selected are retained at the frequency at which they are screened.

(2) Grouping and Screening Criteria

The following grouping and screening criteria was used.

I. Hazard grouping by Denomination

These are hazards that are bounded by the meaning of other hazards. Grouping these hazards together eliminates hazard repetitiveness and captures a single point hazard.

II. Hazard grouping by Plant Effect

As with Criterion I, hazards can be grouped with other hazards based on the effect on the plant. If the hazard's potential impacts for plant are similar, these hazards are grouped to avoid repeated description of definition and protection.

III. Screening by Frequency of Occurrence

These are defined as having a probability of occurrence less frequent than 1 percent of the CDF.

IV. Screening by Site Significance

The list was filtered by identifying those hazards that are explicitly relevant to the site. This criterion is not applicable for GDA. It could be used post GDA phase.

V. Screening by Consequence

A hazard may be screened out based on its potential impact to safety.

(3) Grouping Results

(a) Single Hazards

Internal Hazard Identification [Chapter 7 of PCSR Rev. C] [Ref-25.91] resulted in a comprehensive list of potential Internal Hazards\Internal Hazards groups. This grouped hazards list had been reviewed, amended and expanded with the hazards found in [Ref-25.95]. This resulted in the updated list. The updated list of grouped hazards, that forms the basis of the generic and site-specific internal hazard characterisations and that is also sufficient to support the PSA can be found in the first two columns of Table 25.10.1-1 below. Column 1 gives the hazard group (Note: "hazard group" used in this Section is equivalent to the term "fault group" used in the other Sections.) and the second column lists which individual hazards from the total list are also part of this group.

The third column presents the results of the qualitative screening process: whether or not the hazard needs to be considered in the PSA.

(b) Combinations of internal hazards

Identification of combinations of internal hazards that need consideration in the prioritisation process at this stage was not effective, not even on a general level. The more opportune moment was after the first qualitative screening step, where it was possible that complete hazards would be screened out (see Subsection 25.10.1.5(3)).

Table 25.10.1-1 Grouped and Screened Internal Hazards Overview (1/4)

Hazard group	Internal hazard	Screening result	
Internal Fire		**In**	**Separate PSA on internal fire**
	Fire from Cask Transporter, on-site materials	In	
	Internal fire spreading from other units on the site	NA	Single unit site
	Transportation accidents (effect of transported goods)	In	
	Explosive Electrical Faults	In	
Internal Flooding		**In**	**Separate PSA on internal flooding**
	Pipe failure effects	In	
	Spray	In	
	Steam Release	In	
	Flooding (from fire suppression systems)	In	
	Water currents during a flood	In	
	Snow melt	Out	External hazard
	Ground washout	In	
	Internal flood and harsh environment spreading from other units on the site	NA	Single unit site
	Transportation accidents (effect of transported goods)	In	
Internal Explosion (chemical)		**In**	**Included in internal fire PSA**
	Pipe failure effects	In	
	Transportation accidents	In	
	Pipeline accident (gas, etc.)	In	
	Shockwave from transformer explosion	In	
	Methane	In	
	Explosion within the site (transport)	In	

Table 25.10.1-1 Grouped and Screened Internal Hazards Overview (2/4)

Hazard group	Internal hazard	Screening result	
	Explosion after pipeline accident within the site	In	
	Transportation accidents (effect of transported goods)	In	
Internal Missiles		**In**	
	Turbine missiles	In	
	Disruptive failure of rotating machinery or other equipment	In	
	Missiles from other units on the site	NA	Single unit site
Toxic, corrosive material		**In**	
	Pipe failure effects	In	
	Transportation accidents	In	
	Pipeline accident (gas, etc.)	In	
	Chemical (= toxic) release within the site (transport)	In	
	Chemical (= toxic) release after pipeline accident within the site	In	
Electromagnetic Interference (EMI)		**Out**	**Within design base of plant**
Transport accident (Vehicular impact)		**Out**	• **Within design of plant** • **Included in the definition of another hazard: hazards: fire, explosion, flooding, and toxic hazards.**
	Transportation accidents (effect of transported goods)	Out	Included in the definition of another hazard: hazards: fire, explosion, flooding, and toxic hazards
	Direct impact of heavy transportation within the site	Out	Within design of plant

Table 25.10.1-1 Grouped and Screened Internal Hazards Overview (3/4)

Hazard group	Internal hazard	Screening result	
Pipe Whip (HELB)		**In**	
	Pipe failure effects (see also internal flooding, blast)	In	
	Jet Effects	In	
Blast (= pressure wave not a chemical explosion), building internal		**In**	
Vibration		**Out**	• **Included in the definition of another hazard: pipe whip, missiles, flooding)** • **Within the design base**
Dropped loads		**In**	
	Collapsing or falling loads	In	
	Structural collapse	NA	
Static electricity		**Out**	**Within design base of plant**
Biological agents		**EH**	**Classified as external hazards** **Additionally:** • **Within design of plant** • **Included in the definition of another hazard: internal flooding, HELB.**
	Biological intrusions (accumulations of)	Out	External hazard
	Biological growth	Out	External hazard
	Organic matter (such as leaves)	Out	External hazard
	Micro-biological corrosion	Out	External hazard
Wildlife		**EH**	**Classified as external hazards** **Additionally:** **within design of plant**
	Birds	Out	External hazard

Table 25.10.1-1 Grouped and Screened Internal Hazards Overview (4/4)

Hazard group	Internal hazard	Screening result	
	Insects	Out	External hazard
	Rodent infestation	Out	External hazard
Excavation work within the site area / construction work		**Out**	**Excavation work is out of scope**
Multi-unit issues		**Out**	**Single unit site**

25.10.1.5 Qualitative Screening

The first step in the qualitative screening process is to screen the complete set of hazards for the entire site. The second step is to screen the hazard by location: first on building level and then on safety division / room level.

(1) Screening of Complete Single Hazards

The last column "Screening Result" of Table 25.10.1-1 shows the results of the screening on the hazard group level. Column 2 gives the characterisation of the hazard group. Subsection 25.10.1.4 (3) (combinations of hazards) gives more details about the characterisation. Most of the hazards did not meet the screening criteria and are retained for further analysis.
The hazards that did meet the screening criteria are briefly discussed below.

(a) EMI/RFI: Screened out (within design of plant)

Electromagnetic Interference (EMI) and / or Radio Frequency Interference (RFI) are introduced by switching on or off large electrical loads or by radio equipment such as mobile phones, WIFI equipment, and hand held communicators.
Modern equipment meeting the appropriate standards is resistant against EMI /RFI. It is good engineering practice to install safety related equipment in EMC cabinets, use adequate earthing (grounding) and have restrictions on wireless communication equipment in the direct environment.

(b) Vibration: Screened out by denomination and by consequence (within the design base)

Plant equipment may be susceptible to the effects of vibration resulting from for example rotating equipment. However, low levels of vibration are addressed by the qualification and design requirements of the systems and components. Vibration levels of rotating equipment are limited by design and monitored to prevent failure as result of abnormal vibration. This is further strengthened by an inspection and maintenance programme that minimises the possibility of vibration induced cracking. Next to this, the consequences of a possible vibration induced failure are hazards that are retained for analysis: pipe whip, missiles, flooding etc. Therefore all vibration induced failures by unforeseen circumstances are incorporated as other hazards.

High short term levels of vibration are considered to be bounded by seismic vibration for seismically qualified equipment.

(c) Static electricity: Screened out (within the design basis)

Static electricity can cause electrical equipment, in particular electronic equipment to malfunction (or breakdown). However, it is considered that if modern equipment is designed and installed to conform to the specifications to meet current Electrical and EMI/EMC standards the equipment must be resistant to static electrical effects. To prevent a possible static discharge, it is good engineering practice to take precautions like anti-static floors, ground bracelets and separate earthing as defined in international standards such as the IEC 61000 series.

(d) Biological agents: Screened out partly and partly classified as external hazards.

The following items are considered to be included in the Biological Agents description:

(i) Biological intrusions (accumulations of)
(ii) Biological growth
(iii) Organic matter (such as leaves)
(iv) Micro-biological corrosion

For the items 1), 2) and 3) standard design measures like screens, filters and water treatment, as well as plant design, plant monitoring and maintenance will be implemented to prevent biological agents resulting in system failure. These hazards are therefore screened out as being within the design of the plant. They could also be dispositioned as external hazards and as such screened-out from the internal hazard analysis.
Micro-biological corrosion can occur particularly in static water systems containing untreated water. However, any resulting leakage is bounded by the flooding hazard and, in case of high energy systems, by pipe whip, spray, jet etc. It is a cause of these hazards. This hazard is therefore screened out.

(e) Wildlife: Screened out partly and partly classified as external hazards.

The internal hazard 'Wildlife' considers the hazards resulting from the presence of wildlife infestation resulting in damage to equipment, non-availability of essential plant safety equipment or blockage of ventilation ducts, etc...
The following items are considered to be included in the Wildlife description:

(i) Birds
(ii) Insects
(iii) Rodent infestation

This list is not complete. All animal life forms should be taken into account within this group, e.g. jelly fish, fish, clams, mussels etc... No specific design measures are considered necessary for these potential hazards (apart from good engineering practice, e.g. segregation of air intakes, provision of armoured cables, etc.).

(f) Excavation work within the site area / construction work: Out of scope

This hazard deals with work performed after commissioning of the plant. As the risk cannot be assessed before information on the excavation work to be performed is available, the hazard is screened out from the PSA. To assess the risk of this kind of activity an ad hoc safety case should be built. The future licensee would have to assess the effects of any design change to the plant during operation prior to the implementation of such change and justify that no significant impacts on the safety case of the existing plant results from construction activities.

(g) Transportation accidents: Screened out partly and partly analysed in other hazards

There are numerous transport movements of heavy loads on a nuclear site. Impacts between transport vehicles and safety classified buildings and external structures are in principle possible. Building/structure design and strict regulations on vehicle movement make serious safety consequences improbable.
The hazards of any transported goods/materials are assessed within the hazards: fire, explosion, flooding, and toxic hazards.

(h) Multi-unit issues: Not applicable and therefore screened out

Hazards and hazard consequences in one unit that could affect one or more other units at the same site are referred to as multi-unit hazards. Depending on the specific definition of "site", these hazards could be classified as external or internal hazards. In any case they will only be part of a multi-unit PSA and are as such not included in this report as the present licensing process only considers one unit at the site.

(2) Characterisation of Retained Hazard Groups

(a) Internal Fire

This hazard group includes the fires that start inside buildings or structures. Part of the fire hazard assessment is also explosive electrical faults and high energy arcing faults causing fire. Internal fire is analysed together with Internal Explosion.

(b) Internal Flooding

The hazard group Internal Flooding encompasses all fluid containing systems or components. It is not restricted to water/steam systems alone, or to direct plant systems. Fire suppression, heating and public water supply are potential sources as well. These systems or components can be fixed (such as piping, storage tanks vessels, fire suppression system) or temporarily such as diesel fuel transported on site. Flooding can mean submersion but also spray from a fluid or spray / jet from a (hot) gas like steam. All will have their specific impact on their environment.

(c) Internal Explosion (chemical)

The hazard group Internal Explosion can be seen as sub group of Internal Fire with a specific (additional) consequence. It concentrates on the presence of substances that can explode as result of a chemical reaction. In most cases it is treated and analysed as part of the internal fire analysis. Failure of a high-pressure system creating a pressure wave is treated under the hazard group Blast.
Sources of explosive substances can be fixed (lubricating oil and jacking oil storage tanks, main transformers, hydrogen storage) or temporarily (diesel fuel transport). Consequences to consider are heat radiation, toxicity/asphyxiation, pressure wave and missiles.

(d) Internal Missiles

The group Internal Missiles is restricted to missiles ejected by large rotating machinery such as a steam turbine (blades or a failed rotor), main pumps and the generator. Emphasis for selecting equipment for this group is on the missile potential, not so much on the pressure wave, although this needs off course to be considered in the consequence analysis as well as possible flooding.

(e) Toxic, Corrosive Material

Toxic and asphyxiate substances (fluids and gasses) are part of this group. From a nuclear safety and PSA point of view the impact of Toxic Material narrows the hazard down to the impact of a release on personnel in the MCR and possibly for field operators, whose actions are credited in the PSA. Systems and components are not affected.

(f) Pipe Whip (HELB)

The hazard group Pipe Whip (HELB) encompasses high energy systems (e.g., steam, water, nitrogen, air) with the emphasis not so much on the possible consequences caused by the substance contained by the components but by the dynamic effects of the components: pipe whip and missiles. High energy systems are systems at fluid temperatures over 95°C and a pressure greater than 1,900 kPa or gas containing systems with a pressure over 1,900 kPa. The whipping pipe can impact structures and structures and systems.
The emphasis on the effects of the substance enclosed is within the hazard groups Blast, Internal Explosion and Internal Flooding. Nevertheless, secondary effects as jet impingement, missiles spray and flooding should also be considered.

(g) Blast

The hazard Blast is defined as a pressure wave caused by the sudden release of a medium caused by failure of pressure boundary. This failure is not caused by an explosion of the medium. Systems to consider are high energy systems that contain steam/water, nitrogen, air).

(h) Dropped Loads

Dropped loads can be caused by the failure of (overhead/polar) cranes, lifting rigs and slinging faults. The potential consequences to consider are related to the possible damage to the load, the impact of the load on other SSCs and the structural collapse.

(3) Combinations of hazards

Based on the hazards identified in Table 25.10.1-1, 15 hazards were systematically analysed for possible combinations by reviewing every combination of two hazards [Ref-25.94]. In total 225 combinations were examined and classified according to the coupling mechanisms:

(1) Consequential;
(2) Correlated;
(3) Independent: by chance.

As a result of review on combination, an updated combination table has been developed in Table 25.10.1-2. The crosses in Table 25.10.1-2 below indicate which consequential hazards should be considered, given the primary hazard.

Table 25.10.1-2 Combinations of Internal Hazards to Consider in a Consequence Assessment of a Primary Hazard

Primary Hazard \ Consequential Hazards	Internal Fire	Internal Flooding	Internal Explosion (chemical)	Internal Missiles	Toxic, corrosive material	Pipe Whip (HELB)	Blast (not chemical explosion)	Dropped Loads
Internal Fire	x	x	x	x	x	x	x	x
Internal Flooding	x				x			x
Internal Explosion (chemical)	x	x	x	x	x	x	x	x
Internal Missiles	x	x	x	x	x	x	x	x
Toxic, corrosive material								
Pipe Whip (HELB)	x	x	x	x	x	x	x	x
Blast (not chemical explosion)	x	x	x	x	x	x	x	x
Dropped Loads	x	x	x	x	x	x	x	

Table 25.10.1-2 can be used as guidance in the quantitative steps of the prioritisation process, when the consequences of an initiating hazard have to be translated into loss of safety relevant SSC in order to estimate and model the impact of the initiating event on the plant. The identification of consequential hazards is described as follows:

(a) EMI/RFI: Screened out (within design of plant)

Electro Magnetic Interference (EMI) and / or Radio Frequency Interference (RFI) are introduced by switching on or off large electrical loads or by radio equipment such as mobile phones, WIFI equipment, and hand held communicators.
Modern equipment meeting the appropriate standards is resistant against EMI /RFI. It is good engineering practice to install safety related equipment in EMC cabinets, use adequate earthing (grounding) and have restrictions on wireless communication equipment in the direct environment.

(b) Vibration: Screened out by denomination and by consequence (within the design base)

Plant equipment may be susceptible to the effects of vibration resulting from for example rotating equipment. However, low levels of vibration are addressed by the qualification and design requirements of the systems and components. Vibration levels of rotating equipment are limited by design and monitored to prevent failure as result of abnormal vibration. This is further strengthened by an inspection and maintenance programme that minimises the possibility of vibration induced cracking. Next to this, the consequences of a possible vibration induced failure are hazards that are retained for analysis: pipe whip, missiles, flooding etc. Therefore all vibration induced failures by unforeseen circumstances are incorporated as other hazards.

High short term levels of vibration are considered to be bounded by seismic vibration for seismically qualified equipment.

(c) Static electricity: Screened out (within the design basis)

Static electricity can cause electrical equipment, in particular electronic equipment to malfunction (or breakdown). However, it is considered that if modern equipment is designed and installed to conform to the specifications to meet current Electrical and EMI/EMC standards the equipment must be resistant to static electrical effects. To prevent a possible static discharge, it is good engineering practice to take precautions like anti-static floors, ground bracelets and separate earthing as defined in international standards such as the IEC 61000 series.

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The following items are considered to be included in the Biological Agents description:

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Micro-biological corrosion can occur particularly in static water systems containing untreated water. However, any resulting leakage is bounded by the flooding hazard and, in case of high energy systems, by pipe whip, spray, jet etc. It is a cause of these hazards. This hazard is therefore screened out.

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The following items are considered to be included in the Wildlife description:

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(f) Excavation work within the site area / construction work: Out of scope

This hazard deals with work performed after commissioning of the plant. As the risk cannot be assessed before information on the excavation work to be performed is available, the hazard is screened out from the PSA. To assess the risk of this kind of activity an ad hoc safety case should be built. The future licensee would have to assess the effects of any design change to the plant during operation prior to the implementation of such change and justify that no significant impacts on the safety case of the existing plant results from construction activities.

(g) Transportation accidents: Screened out partly and partly analysed in other hazards

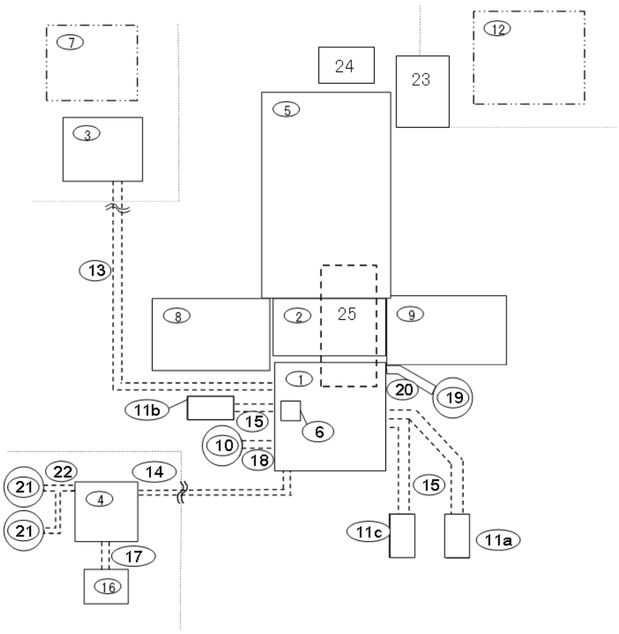
There are numerous transport movements of heavy loads on a nuclear site. Impacts between transport vehicles and safety classified buildings and external structures are in principle possible. Building/structure design and strict regulations on vehicle movement make serious safety consequences improbable.
The hazards of any transported goods/materials are assessed within the hazards: fire, explosion, flooding, and toxic hazards.

(h) Multi-unit issues: Not applicable and therefore screened out

Hazards and hazard consequences in one unit that could affect one or more other units at the same site are referred to as multi-unit hazards. Depending on the specific definition of “site”, these hazards could be classified as external or internal hazards. In any case they will only be part of a multi-unit PSA and are as such not included in this report as the present licensing process only considers one unit at the site.

(4) Limitation of the Number of Buildings to Consider: Scope

The standard UK ABWR plant arrangement is shown in the Figure 25.10.1-1, taken from the chapter 9 (PCSR Chapter 9). The items not in the GDA scope are the Circulating Water Structure (7) and the Spent Fuel Storage Facility (12), which leaves 33 items to be part of the qualitative screening process. The standard plant is located on a site adjacent or close to a body of water with sufficient capacity to provide cooling for all anticipated operational plant states including any accident sequences. Although the location and exact design of the Diverse Additional Generator (DAG) building are still to be decided, the internal events PSA credits this building for long term heat removal (RHR) scenarios. For the present analysis, it is assumed that the design is comparable to one of EDG divisions.



No.	FACILITY
1	REACTOR BUILDING (R/B)
2	CONTROL BUILDING (C/B)
3	HEAT EXCHANGER BUILDING (Hx/B)
4	BACKUP BUILDING (B/B)
5	TURBINE BUILDING (T/B)
6	MAIN STACK
7#	CIRCULATING WATER STRUCTURE (CW/S)
8	RADWASTE BUILDING (Rw/B)
9	SERVICE BUILDING (S/B)
10	CONDENSATE STORAGE TANK (CST)
11	EMERGENCY DIESEL GENERATOR BUILDING (EDG/B)
12#	SPENT FUEL STORAGE FACILITY
13	RCW TUNNEL (R/B-Hx/B CONNECTING SERVICE TUNNEL)
14	R/B-B/B CONNECTING SERVICE TUNNEL
15	R/B-EDG/B CONNECTING SERVICE TUNNEL
16	LIGHT OIL STORAGE TANK (LOT)
17	LOT CONNECTING SERVICE TUNNEL (B/B-LOT)
18	CST CONNECTING SERVICE TUNNEL (R/B-CST)
19	SUPPRESSION POOL TANK (SPT)
20	SPT CONNECTING SERVICE TUNNEL (R/B-SPT)
21	WATER STORAGE TANK FOR B/B
22	B/B-WATER STORAGE TANK CONNECTING SERVICE TUNNEL
23	MAIN, AUXILIARY, AND EXCITATION TRANSFORMERS
24	AUXILIARY/STANDBY TRANSFORMER
25	MAIN STEAM TUNNEL

1 This drawing illustrates conceptual site plan and not intended to show final exact location of the structures.
2 Structures with "#" are NOT included in the GDA scope
3 DAG is not shown, as no decision on location is taken

Figure 25.10.1-1 Generic Site Plan

(5) Limitation of the Number of Buildings to Consider: Safety Relevance

In the first step of the screening process of complete buildings, the buildings in the GDA scope were screened using three criteria shown below:

(a) The hazard causes an IE;
(b) The hazard affects SSC needed in accident mitigation; the building contains safety related SSC;
(c) The hazard in a building has potential to cause criterion a or b in a neighbouring building, when its effects are not contained in the building.

In this step, 4 items were screened out: the service building, the suppression pool tank, the SPT connecting service tunnel, and the DAG building [Ref-25.95]. The buildings and structures retained for further analysis are given in the Table 25.10.1-3.

Table 25.10.1-3 Buildings and Structures Retained for Further Analysis after Screening

Building & Facility No.	Object name
101	Reactor Building
102	Control Building
103	Heat Exchanger Building
104	Rad waste Building
106	Main Stack
107	Back-up Building
108	Turbine Building
110	Emergency Diesel Generator Buildings (3 in number)
502	Light Oil Storage Tank for the B/B (LOT)
505	Condensate Storage Tank, Foundation
515	Water Storage Tank for B/B (2 in number)
506	Main, Auxiliary, And Excitation Transformers (6 in number)
518	Auxiliary/Standby Transformer
601	R/B - EDG/B Connecting Service Tunnels (3 in number)
602	B/B - LOT Connecting Service Tunnel
603	RCW Tunnel
604	R/B - B/B Connecting Service Tunnel
605	R/B - CST Connecting Service Tunnel
618	B/B - Water Storage Tank Connecting Service Tunnels (2 in number)
101/102/108	Main steam tunnel

(6) Presence of Hazards in Buildings

The result of the preceding two steps: screening of complete hazards site wide and screening of complete building without considering hazards is used as input of the next step. This next step in the qualitative screening is to assess per hazard (group) in which building the hazard can occur.

Reasons to screen an item out are:

(a) No hazard is present in the building;

(b) The manifestation of the hazard(s) are limited to the system where the hazard originates and this system and its failure modes are already in the internal events PSA.

The remaining buildings and structures are listed in Table 25.10.1-4.

Table 25.10.1-4 Buildings Retained for Further Analysis after Screening

Building & Facility No.	Object name
101	Reactor Building
102	Control Building
103	Heat Exchanger Building
104	Radwaste Building
107	Back-up Building
108	Turbine Building
110	Emergency Diesel Generator Buildings (3 in number)
506	Main, Auxiliary, And Excitation Transformers (6 in number)
518	Auxiliary/Standby Transformer
601	R/B - EDG/B Connecting Service Tunnels (3 in number)
603	RCW Tunnel
604	R/B - B/B Connecting Service Tunnel
101/102/108	Main Steam Tunnel

The buildings that were screened out at building level are [Ref-25.95]:

- the Main Stack (building 106);
- the Light Oil Tank for the B/B (502);
- the Condensate Storage Tank (505);
- the two Water Storage Tanks for B/B (515);
- the B/B – LOT Connecting Service Tunnel (602);
- the R/B –CST Connecting Service Tunnel (605). and;
- the two B/B-Water Storage Tank Connecting Service Tunnels (618).

(7) Hazards already under assessment

The internal fire and internal flooding are already assessed separately. Buildings/divisions containing only those two hazards could also be screened out from further assessment. The following buildings/structures, with only those two hazards were identified. These buildings are listed in Table 25.10.1-5.

Table 25.10.1-5 Buildings / Structures That Need Only Fire and Flooding Analysis

Building & Facility No.	Object name
506	Main, Auxiliary, And Excitation Transformers (6 in number)
518	Auxiliary/Standby Transformer
601	R/B - EDG/B Connecting Service Tunnels (3 in number)
603	RCW Tunnel
604	R/B - B/B Connecting Service Tunnel
101/102/108	Main steam tunnel

25.10.1.6 Quantitative Screening for at Power

The quantitative screening was only performed for the at Power Plant Operational State in GDA phase. The normal shutdown systems through to the cold shutdown plant operating state were assessed as part of the accident sequence for the internal events Shutdown PSA and were also considered in the at Power Fire, Flood, and Seismic PSAs. But the quantitative screening process will be expanded for shutdown condition post GDA phase because Hitachi-GE considers that an assessment will not provide meaningful information due to lack of detail information on shutdown conditions. In this regard, grouping of hazard for at power condition is considered to be applicable for shutdown condition and the starting point for assessment of shutdown condition. Moreover, all buildings identified for at power condition are retained for shutdown condition and SSCs that are required for shutdown condition are almost same as those required for at power condition without FPC, MUWC and SPCU.

(1) Approach

For the prioritisation task (quantitative screening) the internal events PSA model needs to be used to determine the potential impacts of the hazards: i.e. Conditional Core Damage Probabilities (CCDPs) are calculated for expected loss of buildings, loss of safety divisions etc. and all consequential events that results from the hazard. In combination with the respective initiating frequency for the hazard, risk significant hazards scenarios were identified and prioritised for further, more detailed assessment. The quantitative contributions of the hazards not prioritised for further analysis were retained as risk contributors at their screening value.
For the purposes of a simplified quantitative assessment of the risk resulting from a specific internal hazard, or for the screening of enclosed plant areas, the core damage frequency could be estimated without a detailed Level 1 PSA model for internal hazards. In this case, the general formula for calculating the cumulative contribution to core damage frequency from the specific internal hazard is:

$$f_{hazard\ core\ damage} = \sum_i f_{hazard\ in\ plant\ area\ i} * CCDP_i$$

where:

$f_{\text{hazard core damage}}$ is	the contribution from the specific internal hazard to the core damage frequency;
$f_{\text{hazard in plant area i}}$ is	the frequency of occurrence of the specific internal hazard in plant area '*i*';
$CCDP_i$ is	the conditional core damage probability for plant area '*i*', estimated using the Level 1 PSA for internal initiating events, adapted with conservative assumptions in accordance with the effect in the plant area '*i*' of the internal hazard.

(2) CCDP calculations

For the building or structures retained for quantitative screening the Conditional Core Damage Probability (CCDP) is calculated with the at power PSA model. The approach consists of four distinct steps.

(i) Identification of SSCs impacted from Loss of Building or Divisional Areas
(ii) Selection of basic events
(iii) Selection of Representative Initiating Events
(iv) CCDP calculation

The SSCs for safe shutdown were not explicitly identified for each separate hazard, but the hazard impacts were represented in the analysis case, and the CCDP provided a reasonable estimate of the conditional risk from a hazard.
The Radwaste Building is a special case. It plays no role in the PSA for the reactor. It is retained for assessment to support the non-reactor fault assessment (dose release assessment from Rw/B).

(a) Identification of SSCs Impacted from Loss of Building or Divisional Areas

In order to simulate the loss of a building or divisional area the location (building and division) of all SSC's modelled in the Internal Events At Power PSA (IEAP PSA) was specified using a structured process and making use of the available information to the extent possible (room data sheets, fire area drawings, Japanese ABWR design, etc.).
The SSCs that are required for safe shutdown from the hazards were the same SSCs that are required to achieve safe shutdown for the representative IEs. The SSCs for safe shutdown were not explicitly identified for each hazard, but the hazard impacts were represented in the analysis case, and the CCDP provided a reasonable bounding estimate of the conditional risk from a hazard, as the maximum damage (a total loss of the building/division) was assumed, while in practice the damage might be limited to a room.

(b) Selection of basic events

Based on the components to building/division mapping, the specific basic events given loss of a building / division were identified in the PSA database. These basic events needed in principle to be set to "TRUE" (which equals a guaranteed failure state). However, the following basic events were treated differently:

(i) Test and maintenance events: these events are not impacted by a hazard and there value is not changed;
(ii) Pre-initiator human failure events: these events are not impacted by a hazard and there value is not changed

(iii) Post-initiator human failure events: impacts on post-initiator operation by hazards are already considered by the failures of SSCs which should have been operated or should have supported the operation. Exceptions are post-initiator human failure events that are performed from within the MCR in case the MCR (and/or crew) is lost.

(iv) CCF events: CCF events are treated as events not impacted by a hazard, with the following main exception: common cause failure of redundant components present in the same building or division is considered by the impacts on the individual components e.g., loss of redundant pumps in the same divisional is captured by the failures of the individual pumps.

In the power supply of components and the C&I not all basic events were set to "failed". Per building selected components in the power supply and C&I were chosen in such a way that their failure would make the power supply or C&I chain unavailable. In those cases where, for example, the failure of the pump itself had the same effect as failure of its power supply; there was no need to set any basic events of the power supply to the failed state.

(c) Selection of representative initiating events

To calculate a CCDP for a hazard on building or divisional area level with the IEAP PSA model, an initiating event from this model needed to be selected that matches the impact of the hazard the best or represents an enveloping impact.
The conditional core damage probability (CCDP) analysis support quantitative screening analysis was performed based on the Internal Events At Power (IEAP) Level 1 PSA details:

The LOOP initiator was analysed as a bounding case. In case of internal hazards LOOP were generally not expected, with the exception for the loss of the turbine building.
For most of the buildings/structures this means that a manual shutdown is the most appropriate initiating event, because in most cases the loss of the building concerns systems not needed for the primary processes or systems with enough redundancy to avoid an automatic shutdown. For cases affecting long term LOOP or hazards impacting the turbine building the hazard may be represented by "LOOP > 14 hours" (unrecoverable LOOP). All turbine building systems (condensate, FW etc.) are assumed to be unavailable for long term LOOP events.
Table 25.10.1-6 gives an overview of the buildings that were screened, the CCDP given loss of the building, the initiating event used.

Table 25.10.1-6 Conditional Core Damage Probability for Loss of Specific Buildings / Divisions (1/2)

Building or Division	Building No.	Representative IE	CCDP given building loss	Note
R/B Div.1	101	Manual shutdown	2.80E-09	
R/B Div.2	101	Manual shutdown	6.00E-07	
R/B Div.3	101	Manual shutdown	3.80E-09	
R/B Div.4	101	Manual shutdown	2.00E-09	
MS tunnel	101/102/108	Feedwater line A break outside containment	2.2E-06*	
C/B Div.1	102	Manual shutdown	1.90E-09	
C/B Div.2	102	Manual shutdown	2.10E-09	
C/B Div.3	102	Manual shutdown	1.5E-07	
C/B Div.4	102	Manual shutdown	1.80E-09	
C/B MCR	102	General transient w/o operator actions within 1 h or only possible in MCR	2.40E-06	
Hx/B Div.1 RCW tunnel Div.1	103 603	Manual shutdown	9.10E-07	
Hx/B Div.2 RCW tunnel Div.2	103 603	Manual shutdown	2.90E-09	
Hx/B Div.3	103	Manual shutdown	2.60E-09	
RCW tunnel Div.3	603			
Rw/B	104	Manual shutdown	1.80E-09	
B/B	107	Manual shutdown	1.30E-06	
T/B	108	Feedwater line A break outside containment (with LOOP)	8.7E-06	

Table 25.10.1-6 Conditional Core Damage Probability for Loss of Specific Buildings / Divisions (2/2)

Building or Division	Building No.	Representative IE	CCDP given building loss	Note
EDG/B A R/B - EDG/B tunnel A	110 601	Manual shutdown	4.60E-09	
EDG/B B R/B - EDG/B tunnel B	110 601	Manual shutdown	2.80E-09	
EDG/B C R/B - EDG/B tunnel C	110 601	Manual shutdown	2.50E-09	
R/B – B/B tunnel	604	Manual shutdown	1.30E-06	
Main, Auxiliary, And Excitation Transformers	506	General transient	Not calculated	
Reserve Auxiliary Transformer	518	Manual shutdown	Not calculated	

*: CCDP was calculated given feedwater line A "break". Failure probability to isolate the feedwater A line was included in the CCDP

(3) CCDP Screening Results

(a) Screening Criterion

"*If the hazard affects, directly and indirectly, only components in a single system, AND it can be shown that the product of the frequency of the hazard and the probability of SSC failure given the hazard is at least two orders of magnitude lower than the product of the non-hazard (i.e., internal events) frequency for the corresponding initiating event in the PSA, and the random (non- hazard) failure probability of the same SSCs that are assumed failed by the hazard.*

If the hazard impacts multiple systems, directly or indirectly, DO NOT screen on this basis."

The US NRC Screening criterion is based on a system being failed. Application of this criterion depends on how a system is defined. A system can consist of multiple redundant pumps within a train and have multiple trains performing a function such as high pressure injection. A system failure could therefore result in a specific safety function not being available.

In the UK ABWR the safety systems have been structured to provide redundancy through three separate safety divisions, with segregated high pressure, low pressure, electrical and C&I divisions supporting those systems. The plant design was developed using this principle and the primary argument of the safety case was that it demonstrated that there is sufficient protection against potential faults that could compromise the integrity of barriers separating the divisions so that loss of multiple divisions is not likely. It is therefore appropriate to use the loss of a division instead of loss of a system for the screening criterion. Although losing a division could mean that more than one safety system is affected, given the threefold split of safety divisions of every safety system affected two out of three trains stay available. The criteria were not written to retain hazards that impact multiple systems within the same division. If a hazard is shown to impact an area, the entire division is assumed lost, which means all systems in the division are lost. It is therefore appropriate to use the loss of a division instead of loss of a system for the screening criteria.

(b) Hazards retained from Qualitative Screening

The following hazards are retained after the qualitative screening step.

Internal fire	Analysis of internal fire is deferred to Internal Fire PSA
Internal flooding	Analysis of internal flood is deferred to Internal Flood PSA
Pipe whip and Jet impingement	Impacts included in the Internal Flood PSA
Explosion (chemical)	Impacts included in the Internal Fire PSA
Dropped loads	Retained, Heavy load drops deferred to Heavy load drop assessment
Internal missile	Retained for quantitative analysis
Internal blast	Retained for quantitative analysis
Toxic substances	Retained for quantitative analysis

Internal Fire was analysed plant wide in the Internal Fire PSA for at power operation and representative shutdown plant operating states (see Section 25.10.2). This hazard can therefore be excluded from this analysis

As with Internal Fire, Internal Flood was analysed plant wide in the Internal Fire PSA for at power operation and representative shutdown plant operating states (see Section 25.10.3). This hazard can therefore be excluded from this analysis

This means that in the following screening analysis the focus is on the hazards Dropped Loads, Internal Missiles, and Internal Blast. From a PSA point of view the toxic hazard narrows down to the impact of a release on personnel in the MCR and possibly for field operators, whose actions are credited in the PSA. (see Section 25.10.1.5 (6) and [Ref-25.95] for a full list of field actions)

25.10.1.7 Conclusion

Both qualitative screening and quantitative screening were conducted and analysis of internal fire and flood are mainly required to be performed. Thus, Internal Fire PSA and Internal Flood PSA were conducted and the results of these assessments are described in Sections 25.10.2 and 25.10.3. Internal Fire including Explosion (chemical) Impacts effect was analysed plant wide in the Internal Fire PSA for at power operation and Internal Flood including Pipe whip and Impacts effect is analysed plant wide in the Internal Flood PSA. All other internal hazard retained after screening will be analyse in the post GDA phase.

25.10.2 Fire PSA

This section presents a summary of the Internal Fire PSA.

25.10.2.1 Scope

The scope of the Internal Fire PSA for GDA is as follows:

Hazard Location and Sources

- The analysis examines the potential risk from fire events originating anywhere within the site boundary: Fire events caused by fixed equipment as well as transient sources (e.g. welding and cutting, including fires involving the flammable / combustible gases and liquids, are considered. The effects of thermal damage, high energy arcing faults and smoke are addressed.

- Consideration is given to fires in the switchyard to ensure that the risk from such fires has been adequately addressed in the Internal Events analysis. Should the review of the basis of the internal events loss of offsite power frequency and recovery probability related to Switchyard faults not address fire impacts, such events are explicitly considered within the internal Fire PSA [Ref-25.96].

Coincident Events

- A quantitative basis for excluding other initiating events coincident with the internal hazards, including fire, is included in the analysis.

- The analysis of risk from seismically induced fire events is addressed qualitatively in accordance with NUREG/CR-6850 [Ref-25.96], Chapter 13. Insights from the Seismic PSA (section 25.11.2) work supported this effort.

Plant Operating States (POSs)

- Core Damage Frequency (CDF) and Large Release Frequency (LRF) were quantified for the at Power state (including full power and low power).

- Fuel Damage Frequency (FDF) in the reactor during shutdown was quantified as a scoping analysis for representative POSs to demonstrate that the risk from Internal Fire during shutdown is insignificant.

Sources of Radioactive Release

- Accidents related to the potential release of radioactive material from the reactor vessel and containment was addressed within the Internal Fire PSA. In order to achieve this, it was necessary to propagate the Internal Fire analysis through both the Level 1 and Level 2 analyses including any fire related impacts which impact the containment integrity directly or systems credited for ensuring the protection of its integrity.

- Release from the SFP was also considered. FDF at Power (representative POS) was quantified as a scoping analysis to demonstrate that the risk from Internal Fire on SFP is insignificant.

- Releases from the radioactive waste management system were excluded from the Internal Fire PSA.
- Fresh fuel in the fuel route was evaluated qualitatively to demonstrate that sub-criticality is maintained due to physical separation.
- Spent fuel in the fuel route was evaluated qualitatively to demonstrate that sub-criticality is maintained due to physical separation and the consequences of radioactive release are small using passive protection mechanisms.

25.10.2.2 Methodology

The Internal Fire PSA for UK ABWR GDA was performed based on the guidance provided in NUREG/CR-6850 [Ref-25.96] and incorporates more recent updates to the guidance provided through the US NRC/industry FAQs (Frequently Asked Questions) process, issued by the Nuclear Energy Institute. The full NUREG/CR-6850 methodology [Ref-25.96] and ASME /ANS PSA Standard [Ref-25.97] were reviewed.

The availability of the plant design details limits the extent that the Internal Fire PSA can be performed during the GDA timescales. For design information not available in the GDA timescales, assumptions regarding the plant design and operation were made with reference to existing ABWR plants designs and procedures.

Figure 25.10.2-1 gives an overview of the Internal Fire PSA process and is referenced directly from NUREG/CR-6850 [Ref-25.96]. The detailed methodology for each task is described below.

Task 1: Plant Boundary and Partitioning

This task defines the GPAB (Global Plant Analysis Boundary) and subdivides the GPAB into fire compartments (Physical Analysis Units: PAUs) to be used in the analysis. Initially, the Fire PSA uses one fire scenario per compartment where all equipment within that compartment is assumed to be lost (Whole Room Damage: WRD). The fire compartments consist of one or more rooms. The room information, including the spatial relationship and penetrations to adjoining rooms, is added to the Fire PSA database. For the UK ABWR, the GPAB was limited to the GDA scope of buildings identified in [Chapter 9 of PCSR], which includes the main buildings (e.g. Reactor, Control, Turbine, Heat Exchanger, and Backup Building) and their connecting service tunnels.

All the PAU definitions for the at Power analysis including Primary Containment Vessel (PCV) compartment were retained for the scoping analysis of Shutdown POSs. Degraded boundaries due to maintenance periods were considered in Task 11c (Multi Compartment Analysis).

Task 2: Component Selection

This task defines the equipment list and associated fire-induced failure modes. The equipment selected are those that, if fire damaged, can lead to an initiating event or are required to respond to an initiating event and bring the plant to a safe and stable state. This task also defines the potential fire induced initiating events and corresponding accident sequences to be used in the development of the Internal Fire PSA model (this model development activity is performed in Task 5). The equipment selection also identifies instrumentation whose failure, including spurious operation, would impact the reliability of operator actions. Furthermore, the potential for multiple spurious operations due to fire is analysed at this stage and additional equipment are identified.

Task 3 and 9: Cable Selection and Routing

Based on the equipment identified in Task 2, the Cable Selection/Circuit Failure Analysis identifies and maps the cables required for power or control for each equipment and also maps cable locations to the PAUs. The circuit failure analysis task reviews any additional impacts from fire induced circuit failures (e.g. hot shorts) on the electrical distribution system and postulates any further failures as necessary. Any interlocked circuits and associated equipment are identified and included at this stage. The completion of this task enables the analyst to identify the impact of fire to the equipment in the affected PAU and the impacts from damage to cables located in the PAU. This relationship of fire impact in a given PAU to equipment and associated cables is identified and loaded into the Internal Fire PSA database.

Task 4: Qualitative Screening

This task uses information from the preceding steps, to qualitatively screen PAUs if whole room damage produces no initiating event and no impact to equipment required to mitigate an initiating event. This task reduces the number of fire compartments retained for quantitative analysis. For the UK ABWR, no PAUs were qualitatively screened, as Task 1 was limited to the selection of buildings defined in the GDA scope comprising essentially the main buildings and their connecting service tunnels. All of these structures were assumed to lead to an initiating event or to contain accident-mitigating equipment or cables. Therefore, no qualitative screening was performed.

Task 5: Fire Plant Response Model Development

This task develops the probabilistic risk model used to analyse each identified fire scenario. This task also defines the relationship between the fire impacts (e.g. on equipment, cables and operator actions) and the relevant elements of the probabilistic risk model. The Internal Events PSA is used as the basis for the plant response modelling and is modified to model the fire impacts for each scenario, which relate to equipment loss of function or spurious operation, and instrumentation failures, including spurious operation, that impact operator actions. Multiple spurious operation scenarios (as defined in Task 2) are also added to the risk model. The output is a revised model which is capable of being used to generate risk estimates for each fire scenario based on the postulated initiating event and equipment affected. The model includes the new fault trees and event trees specific to the internal Fire PSA, and the revised fault trees and event trees from the base Internal Events model. This task produces a logic model that can be used by FRANX [Ref-25.37], in conjunction with the fire scenario definition, to generate conditional core damage and large release probabilities for each fire scenario.

Task 6: Fire Ignition Frequency

This task defines the ignition (fire initiating event) frequency for each fire scenario. Generic fire frequencies [Ref-25.98] were apportioned to the ignition sources within each fire compartment that can ignite and lead to a damaging fire. The ignition source inventory comprises fixed ignition and transient ignition sources. For the initial whole PAU damage evaluations, the ignition frequency for each compartment is the sum of the fixed ignition and transient ignition source frequencies. This generates a total ignition frequency for the compartment which is entered into the Fire PSA database. The database contains the qualitative details of the fire impacts for each fire scenario and the initiating event frequency for each scenario.

For the scoping analysis of Shutdown POSs, the generic frequencies specific to shutdown states (provided in [Ref-25.98]) were used. The transient influencing grades were increased for the PAUs under maintenance in specific POSs in order to consider potentially higher transient fire frequencies than those in the PAUs with no maintenance.

Task 7: Quantitative Screening

This task produces an initial quantification of the FRANX model using the initial fire scenario definitions (Tasks 1-4 & 6) and plant response model (Task 5). The FRANX model (which itself is a database comprising a series of linked database tables) is synchronised with the Internal Fire PSA database to create the list of fire scenarios and their impacts with links to the plant response model. FRANX is capable of quantifying each fire scenario individually and also all scenarios simultaneously to generate scenario CDF/LRF and total CDF/LRF respectively. At this stage, it is important to note that the fire scenarios are defined on the basis of whole PAU damage. This means that any ignition source within the PAU leads to a loss of all of the equipment and cables within the compartment. The Task 7 quantification is highly conservative, as not all ignition sources are capable of damaging all equipment within a compartment, due to the different heat release rate profiles of the different types of ignition sources, and distances to targets. The Task 7 quantification also does not take into account any potential suppression actions that might be possible. Task 7 provides an effective means to prioritise those PAUs that receive more detailed analysis (detailed fire scenario development). Using the individual scenario and cumulative quantitative prioritisation criteria, an evaluation can be made on a scenario by scenario basis of whether a scenario requires more detailed evaluation as part of the later tasks or whether it can be retained in the model at its screening value.

Following the outcome of the evaluation against the quantitative screening criteria, a number of scenarios (compartments) were identified for detailed fire modelling or scenario analysis as the existing whole compartment burnup scenario was too conservative. The detailed fire modelling and scenario analysis is carried out as part of Task 11 as described below.

Task 10: Circuit Failure Mode Likelihood Analysis

This task provides a probabilistic assessment of the likelihood that a cable will experience one or more specific fire-induced failure modes resulting in spurious operation of equipment. The spurious operation probabilities are based on a number of factors including component type, cable type and power source. This information is available as an output from Task 2, and is used to develop the spurious operation probabilities for affected components. If these impacts are found to be risk-significant following initial Task 14 quantification, then more detailed circuit failure analysis can be conducted leading to refinement of spurious operation potential. Task 10 also provides analysis for the spurious operation duration probabilities (self-healing events). These events are included in the Internal Fire PSA if components return to a specific position given the cable fault clears, and that component change of state can be credited as a success of the function being modelled.

Task 8 and 11: Detailed Fire Modelling (DFM)

This task consists of three main subtasks, namely intra compartment modelling (Task 11a), Main Control Room analysis (Task 11b) and Multi-Compartment analysis (Task 11c). There is a fourth activity associated with structural steel analysis which is not formally part of Task 11 according to the NUREG/CR-6850 task numbering but is often carried out in conjunction with other Task 11 activities as the inputs to Task 11d are available at the time of other Task 11 activities. This is known as Task 11d for the UK ABWR GDA. Fundamentally the output of Task 11 is a set of refined fire scenarios replacing or supplementing the existing fire scenarios quantified as part of Task 7. The refined fire scenarios are added to the Internal Fire PSA database and ultimately used in FRANX for quantification in Task 14. Further refinements and therefore quantification of the scenarios are possible if it is found that the scenario CDF/LRF values are too limiting and valid means of refinement are still available. Each subtask is described in further detail below.

Task 11a: Single Compartment Analysis

For each scenario surviving quantitative screening, a detailed fire model was developed that identifies and characterises fire growth and target damage likelihood including the following aspects for each compartment: a) fire compartment geometries and penetrations, b) fire detection and suppression, c) ignition source characteristics including applicable heat release rate profile, severity factors, and identification of secondary combustibles, and d) the location of targets. This information is all collated to conduct fire propagation and growth analysis, and used to identify specific fire damage states and associated probabilities including the assessment of the suppression terms. The new scenarios represent multiple damage states ranging from limited damage to target sets through to whole compartment damage.

Task 11b: Main Control Room (MCR) Analysis

The MCR analysis is a fire scenario that requires detailed fire modelling due to the risk significance for whole room burnout. Task 11b shares similarities with Task 11a but with some scenarios requiring calculation of the probability of control room abandonment due to adverse habitability conditions caused by a MCR fire. Additionally, scenarios may include fire conditions which require shutdown from outside the MCR due to loss of adequate instrumentation or control. The prediction of the adverse habitability conditions is conducted using a fire modelling tool such as CFAST which provide times at which the abandonment criteria are reached for various scenarios. The probability of forced abandonment of the control room for each peak heat release rate (HRR) value is calculated as the product of the corresponding severity factor and the probability of failure to suppress the fire before abandonment conditions are reached. This is combined with the failure of the operators to successfully mitigate the fire scenario impacts from a reserve control room or panel to calculate a CCDP or CLRP value. Initially a screening value is used for this operator recovery using alternative shutdown procedures. However, this can be refined with more detailed HRA as part of Task 12 if necessary once the CDF and LRF estimates are generated as part of Task 14. The MCR fire scenario frequencies and the appropriate fire damage definitions are added to the Internal Fire PSA database and used in FRANX in the same way as the other detailed fire modelling scenarios [Ref-25.37]. In addition, the scenarios resulting in MCR abandonment require altered HEPs for those operator actions that can be affected by the operations transfer to the Remote Shutdown System (RSS) or Back-up Building (B/B) control room.

Task 11c: Multi-Compartment Analysis (MCA)

This task consists of a number of steps which first seeks to identify the potential exposing and exposed compartment combinations. This is performed using a matrix providing a listing of connected or adjacent fire compartments. Successive screening criteria are applied that are both qualitative (exposed compartment does not contain equipment considered in the PSA) and quantitative to prioritise the analysis of compartments which contain ignition sources capable of generating fire impacts that can propagate from one compartment to another either through openings or (single) barrier failures. Similar to Task 11a, a characterisation of each compartment, its detection and suppression capabilities, ignition source characteristics etc. is carried out and a set of scenario frequencies are developed for each multi-compartment combination. These scenarios and their associated frequencies are then added to the Internal Fire PSA database and ultimately passed to FRANX in a similar way as it is done for the detailed fire modelling scenarios.

Four types of MCA scenarios were developed.

- Type 1: Potentially impacting temperature sensitive equipment in exposed compartment(s) by Hot Gas Layer (HGL).

- Type 2: Not producing a damaging HGL in the exposing compartment but potentially impacting PSA equipment/cables on the opposite side of a non-rated barrier to a damaging plume temperature or radiant heat flux.

- Type 3: Producing a damaging HGL in the exposing compartment, and conservatively assuming all equipment and cables in the associated Fire Area, surrounded by fire barriers, are damaged (as initial scenario definition).

- Type 4: Associated Type 3 scenario with further impact on adjacent Fire Area with evaluated barrier failure probability.

Task 11d: Fire Impact on Structural Steel

This is a review of fire scenarios which have the potential to generate damaging effects that can damage exposed structural steel. This task uses the existing fire scenario definitions coupled with locations where exposed structural steel is present to identify whether those fire scenarios could lead to conditions where the integrity of the structural steel is challenged. If such scenarios are found, a review is carried out to identify whether the impacts from the collapse of the structural steel are bounded by the existing fire scenario impacts in terms of loss of equipment, cables and impact on operator actions. If the existing scenarios are not bounding, then new scenarios need to be added to the Internal Fire PSA database to capture the frequency and consequence of structural steel collapse. These scenarios and their associated frequencies are then input to FRANX as described above.

Task 12: Human Reliability Analysis

The fire impacts postulated in Tasks 1-3 include not only equipment but also operator actions that are credited as part of the accident mitigation response. The operator actions are those carried over from the Internal Events PSA that are applicable to fire events (as outlined in Task 5) as well as additional actions identified specifically for the Internal Fire PSA that are directed by the fire response procedures or recovery actions that can restore the functions, systems, or components on an as-needed basis. The human error probabilities (HEPs) associated with these operator actions are reviewed with consideration of the potential fire impacts on human performance and to equipment required by the operator to diagnose and respond to a fire induced initiating event. This task generates fire-scenario-specific HEPs for use in the Internal Fire PSA Plant Response Model.

Task 13: Seismic Fire Interactions

This is a qualitative review designed to identify specific scenarios where seismically induced failure of equipment considered in the PSA, can lead to a fire or affect fire detection/suppression and manual firefighting equipment. Given this is a qualitative review and has no impact on the Internal Fire PSA quantification; it is not discussed further here.

Task 14: Internal Fire Risk Quantification

The development of the fire risk quantification is an iterative process involving Tasks 7 through 12. As analysis refinements are developed, they are incorporated into the fire risk model. This task represents final quantification of these scenarios after iteration is complete.

This task represents the integration of all fire scenarios. Each fire scenario is defined by the following characteristics (except for those screened where the relevant factors are retained):

- An ignition source and associated frequency at an assigned heat release rate.

- Probability of occurrence for the associated heat release rate – known as the severity factor.
- Damaged compartment set for the resulting fire (from Task 11 if detailed fire modelling performed, otherwise full compartment burnout assumed with the corresponding fire impacts).
- Success or failure of automatic detection and suppression, and associated probability (from Task 11 results).
- CCDP/CLRP for the associated compartment damage set (CFDP for shutdown states and SFP).
- CDF and LRF (FDF for shutdown states and SFP).

Appropriate cutoff values are established for inclusion of fire scenarios. The cutoff values were identified and documented with an appropriate technical justification for the application.

Task 15: Uncertainty and Sensitivity Analysis

The development of a risk assessment inherently results in uncertainty in the analysis results. Task 15 involves the identification and treatment of uncertainty and sensitivity analysis in the overall Internal Fire risk assessment.

This task consists of identifying and characterizing the sources of parametric and epistemic uncertainty attributed to the Internal Fire PSA model development, by assigning and performing appropriate sensitivity studies for alternate fire parameters and fire models. The sources of uncertainty for each subtask in the Internal Fire PSA are identified and characterised (including whether they are modelling or data uncertainties), and appropriate treatment determined and applied consistent with current industry prevailing good practices. Final results from Task 14 are re-run in the Internal Fire PSA model, using alternate data and models, as appropriate to understand the sensitivity to selected assumptions, modelling and inputs. The changes in CDF/LRF are noted and discussed.

The uncertainty analysis follows the guidance provided in NUREG/CR-6850, Appendix V to:

(1) Systematically identify the uncertainties associated with the inputs to the Internal Fire PSA model.

(2) Develop and justify strategies for addressing each uncertainty (e.g. explicit representation, sensitivity analysis or qualitative treatment only).

(3) Propagate uncertainties through the model.

(4) Provide insights regarding the risk significance of fire scenarios.

In some cases, due to the status of the UK ABWR plant design, data from a surrogate plant was used. Surrogate data was treated as an assumption although justification regarding its applicability to the UK ABWR is provided as far as possible. Where feasible, the sensitivity of the UK ABWR fire induced risk results to the assumptions is addressed in this task.

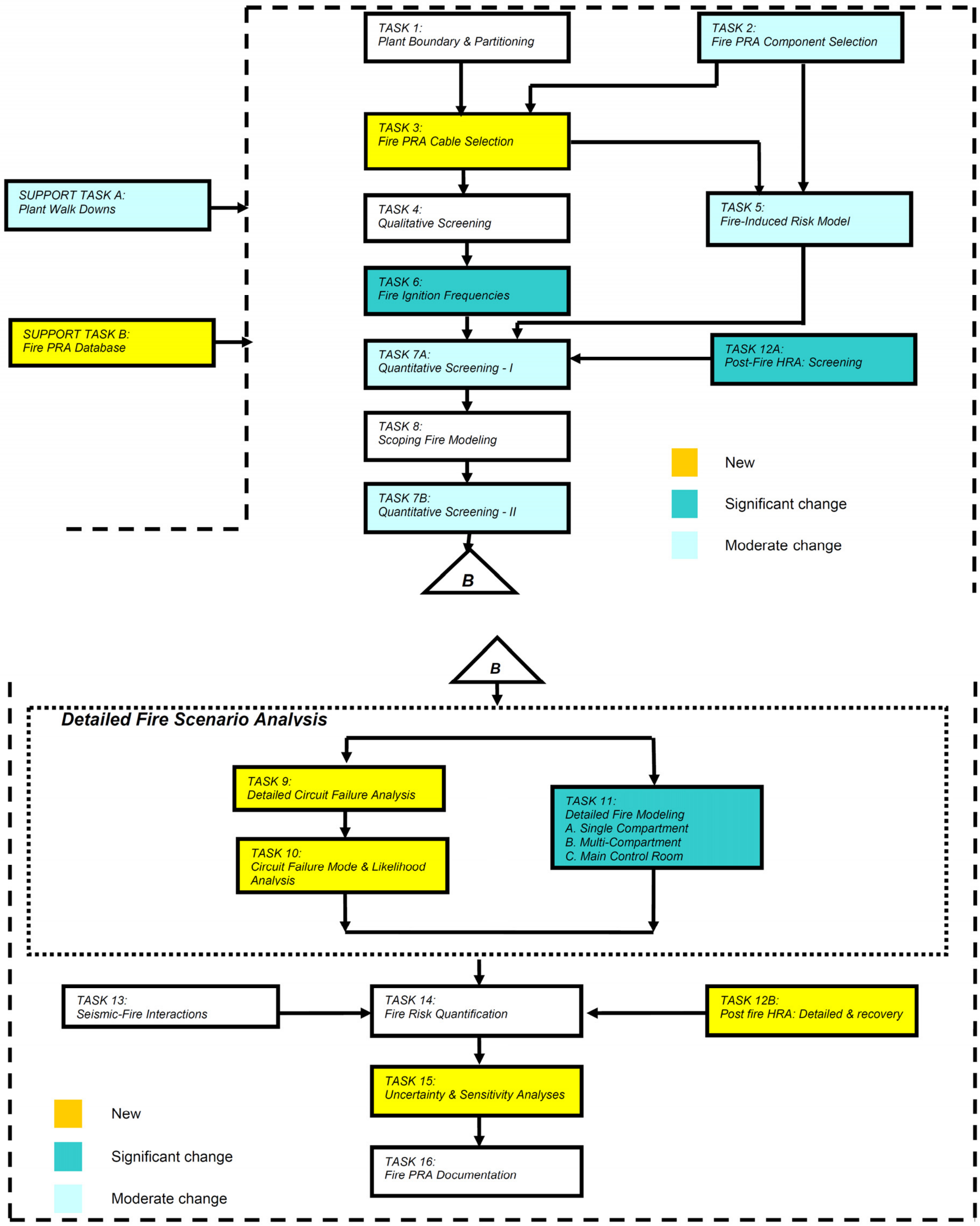


Figure 25.10.2-1 Fire PSA Process Flow in NUREG/CR-6850 [Ref-25.96]

25.10.2.3 Quantification Results

(1) At Power results

The total calculated Internal Fire at Power CDF and LRF results are provided in Table 25.10.2-1 below.

For this calculation, the definition of a significant accident sequence is in agreement with the PRA Standard [Ref-25.97] (excerpted below).

> *"significant accident sequence: one of the set of accident sequences resulting from the analysis of a specific hazard group, defined at the functional or systemic level, that when rank ordered by decreasing frequency, sum to a specified percentage of core damage frequency for the hazard group, or that individually contribute more than a specified percentage of core damage frequency. For this version of the PRA Standard the summed percentage is 95 % and the individual percentage is 1 % of the applicable hazard group".*

The same percentage level was used for LRF. Significant basic events and cutsets were not specifically defined, as they are incorporated within the definition of the fire sequences used in this calculation as below:

> FDS frequency, Fire Scenario Frequency (FSF), addresses the contributors to the damage state, and the CCDP/CLRP calculations are based on cutsets and basic event probabilities. Each contributing factor (fire ignition source, detection, suppression, contributors to CCDP/CLRP) could be included in defining significant cutsets and basic events. This can be performed at some future date, if deemed beneficial. For this Internal Fire PSA, the results available from each of the Task 11 notebooks plus the results provided here are deemed sufficient.

The top 30 fire damage states:

- account for the over 60 percent of the calculated Internal Fire CDF and LRF generated by the Internal Fire scenarios.
- encompass all scenarios individually contributing more than 0.8 percent of the calculated Internal Fire CDF and LRF.

The accident classes contributing greater than 1 percent of the calculated Internal Fire CDF are shown in Figure 25.10.2-2 below. These accident classes contribute approximately 98 percent of the total Internal Fire CDF. The definitions of the accident classes are taken from those for IEAP PSA (Table 25.4.3-2).

Table 25.10.2-2 presents the UK ABWR release category and their contribution to the total Internal Fire LRF. 7 release categories contribute greater than 1 percent of the total Internal Fire LRF. These release categories contribute approximately 99 percent of the total Internal Fire LRF. Those release categories contributing greater than 1 percent of the calculated Internal Fire LRF are shown in Figure 25.10.2-3 below.

LERF was also estimated. Since the total LRF could be significantly overestimated by the non-minimal cutsets and not applying ACUBE, LERF was not calculated by summing up the LRFs of relevant release categories. Instead, the total contribution from the relevant release categories was calculated as 61 percent. This was multiplied to the total LRF (1.64E-06 /y), resulting in the "estimated" LERF of 1.0E-06 /y.

Table 25.10.2-1 Internal Fire at Power CDF/LRF Summary Table

Scenario Group	CDF (/y)	Contribution to Fire at Power CDF	LRF (/y)	Contribution to Fire at Power LRF
Main Control Room	7.38E-10	0.0 %	5.51E-10	0.0 %
Detailed Fire Models*	6.73E-07	35.6 %	6.04E-07	36.9 %
Whole Room Damage**	1.65E-07	8.7 %	1.29E-07	7.9 %
ALL single compartment	8.38E-07	44.3 %	7.34E-07	44.8 %
MCA Type1***	2.72E-09	0.1 %	1.95E-09	0.1 %
MCA Type2***	7.51E-07	39.7 %	7.05E-07	43.1 %
MCA Type3***	9.85E-08	5.2 %	7.52E-08	4.6 %
MCA Type4***	2.01E-07	10.6 %	1.21E-07	7.4 %
ALL multi compartment	1.05E-06	55.7 %	9.03E-07	55.2 %
At Power Total	1.89E-06	-	1.64E-06	-

* Not including MCA scenarios

** WRD scenarios of PAUs which did not receive DFM.

*** MCA scenario “Types” are explained in Section 25.10.2.2 (section of Task 11c).

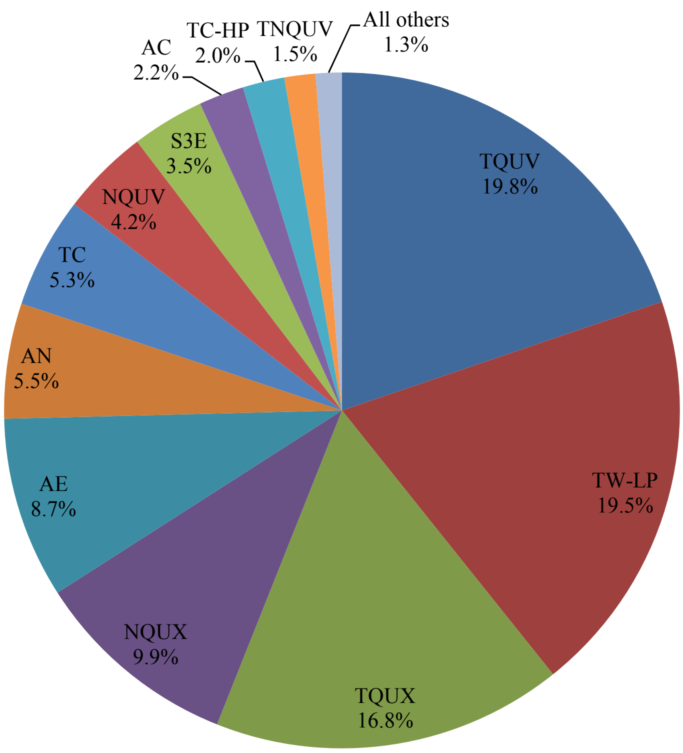


Figure 25.10.2-2 Contribution of Accident Class to Total Internal Fire at Power CDF

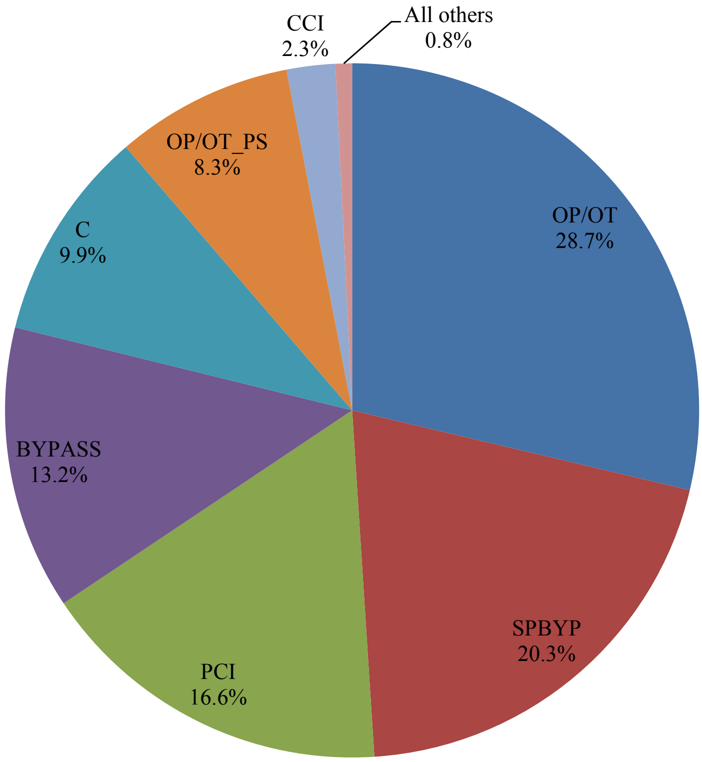


Figure 25.10.2-3 Contribution of Release Category to Total Internal Fire at Power LRF

Table 25.10.2-2 Contributions of Release Categories to Total Internal Fire at Power LRF

Release Category	Containment Failure Mode	Contribution	Inclusion in LERF
Late Containment Failure	OP/OT	28.7 %	N
S/P Bypass	SPBYP	20.3 %	Y
PCV Isolation Failure	PCI	16.6 %	Y
Containment Bypass	BYPASS	13.2 %	Y
Early Containment Failure	C	9.9 %	Y
Late Containment Failure (D/W breach with PCV spray success)	OP/OT_PS	8.3 %	N
MCCI	CCI	2.3 %	N
RPV rupture (with LDF success)	RR_LD	0.5 %	Y
Direct Containment Heating	DCH	0.1 %	Y
Ex-vessel FCI	PE	0.1 %	Y
In-vessel FCI	RE	0.0 %	Y
RPV rupture	RR	0.0 %	Y

(2) Scoping analysis results for shutdown states and SFP

In the scoping analysis, quantification was limited to POS B-2 and POS C. POS C was selected due to the highest risk contribution in the Internal Events Shutdown PSA (see Section 25.8). POS B-2 was included in the unique condition, i.e., SFP is covered due to open pool gate, and less mitigation systems available compared to POS B-1.

The total calculated Internal Fire FDFs during POS B-2 and POS C are provided in Table 25.10.2-3 and Table 25.10.2-4 below.

In the scoping analysis, quantification was limited to POS F (at Power). POS F for SFP PSA was selected due to the longest duration (highest ignition frequency) and also because risk of SFP during shutdown was studied in the scoping analysis of Shutdown POS B-2.

The total calculated Internal Fire FDF at SFP during POS F is provided in Table 25.10.2-5 below.

Table 25.10.2-3 Internal Fire FDF Summary Table for POS B-2

Scenario Group	CDF (/y)	Contribution to POS B-2 Fire FDF
Main Control Room	4.46E-12	0.0 %
Detailed Fire Models*	1.60E-09	8.2 %
Whole Room Damage**	4.98E-09	25.6 %
ALL single compartment	6.58E-09	33.8 %
MCA Type1***	2.30E-11	0.1 %
MCA Type2***	1.10E-08	56.5 %
MCA Type3***	1.55E-09	8.0 %
MCA Type4***	3.01E-10	1.5 %
ALL multi compartment	1.29E-08	66.2 %
POS B-2 Total	1.95E-08	-

* Not including MCA scenarios

** WRD scenarios of PAUs which did not receive DFM.

*** MCA scenario “Types” are explained in Section 25.10.2.2 (section of Task 11c).

Table 25.10.2-4 Internal Fire FDF Summary Table for POS B-2

Scenario Group	CDF (/y)	Contribution to POS C Fire FDF
Main Control Room	2.29E-11	0.0 %
Detailed Fire Models*	4.10E-09	9.3 %
Whole Room Damage**	1.27E-08	28.9 %
ALL single compartment	1.68E-08	38.2 %
MCA Type1***	3.73E-10	0.8 %
MCA Type2***	1.64E-08	37.3 %
MCA Type3***	1.52E-09	3.5 %
MCA Type4***	8.90E-09	20.2 %
ALL multi compartment	2.72E-08	61.8 %
POS C Total	4.40E-08	-

* Not including MCA scenarios

** WRD scenarios of PAUs which did not receive DFM.

*** MCA scenario "Types" are explained in Section 25.10.2.2 (section of Task 11c).

Table 25.10.2-5 Internal Fire FDF Summary Table for SFP POS F

Scenario Group	CDF (/y)	Contribution to SFP POS F Fire FDF
Main Control Room	1.59E-08	2.7 %
Detailed Fire Models*	8.03E-08	13.4 %
Whole Room Damage**	9.92E-08	16.6 %
ALL single compartment	1.95E-07	32.7 %
MCA Type1***	1.79E-09	0.3 %
MCA Type2***	1.81E-07	30.3 %
MCA Type3***	1.09E-07	18.2 %
MCA Type4***	1.11E-07	18.6 %
ALL multi compartment	4.03E-07	67.3 %
SFP POS F Total	5.98E-07	-

* Not including MCA scenarios

** WRD scenarios of PAUs which did not receive DFM.

*** MCA scenario "Types" are explained in Section 25.10.2.2 (section of Task 11c).

(3) At Power refinement results

The significant conservatism in the Internal Fire at Power PSA base case, described in (1) of this section, made the calculated overall risk of UK ABWR generic design conservatively high and impacted the use of PSA to inform the design process for ALARP demonstration.

The Internal Fire at Power PSA was significantly refined from the base case, by applying newly available design information, more detailed analyses and removing conservatisms in the quantification process. The purpose of the refinement is to calculate a more realistic fire risk of UK ABWR in GDA phase, and to make the UK ABWR PSA more useful for the ALARP demonstration. Individual refinements are based on newly available design information and technical considerations to remove conservatisms in the modelling approach.

As the results of various refinements of risk significant scenarios, CDF by fire at Power reduced from 1.89E-06 /y to 4.95E-07 /y. LRF by fire at power reduced from 1.64E-06 /y to 2.65E-07 /y. The total calculated Internal Fire CDF and LRF results at Power are summarised in Table 25.10.2-6 below. The most significant reduction was achieved for Type 2 MCA scenarios, since a number of significant scenarios were screened by applying the updated fire zone drawings. The reductions for Type 3 and Type 4 MCA scenarios were also benefitted mainly from the updated fire zone drawings as well as the refinements of oil spill fires. Reduction of single compartment DFM scenarios was benefitted mainly from the site-specific cabinet layout drawing in B/B as well as the removal of some conservatisms in DFM. Reduction of WRD scenarios was due to an additional DFM for a PAU and removing double-counting of scenario frequencies with associated MCA scenarios.

The accident classes contributing greater than 1 percent of the calculated Internal Fire CDF are shown in Figure 25.10.2-4 below. These accident classes contribute approximately 99 percent of the total Internal Fire CDF. The definitions of the accident classes are taken from those for IEAP PSA (Table 25.4.3-2).

Table 25.10.2-7 presents the UK ABWR release category and their contribution to the total Internal Fire LRF. Four release categories contribute greater than 1 percent of the total Internal Fire LRF. These release categories contribute approximately 98 percent of the total Internal Fire LRF. Those release categories contributing greater than 1 percent of the calculated Internal Fire LRF are shown in Figure 25.10.2-5 below.

LERF was also estimated. Since the total LRF could be significantly overestimated by the non-minimal cutsets and not applying ACUBE, LERF was not calculated by summing up the LRFs of relevant release categories. Instead, the total contribution from the relevant release categories was calculated as 99 percent. This was multiplied to the total LRF (2.65E-07 /y), resulting in the “estimated” LERF of 2.64E-07 /y.

Table 25.10.2-6 Internal Fire at Power Refined CDF/LRF Summary Table

Scenario Group	CDF (/y)	Contribution to Fire at Power CDF	LRF (/y)	Contribution to Fire at Power LRF
Main Control Room	7.82E-09	1.6 %	1.95E-09	0.7 %
Detailed Fire Models*	2.09E-07	42.2 %	1.78E-07	67.2 %
Whole Room Damage**	8.24E-08	16.6 %	3.42E-08	12.9 %
ALL single compartment	2.99E-07	60.4 %	2.14E-07	80.8 %
MCA Type1***	5.38E-09	1.1 %	3.97E-09	1.5 %
MCA Type2***	2.12E-08	4.3 %	6.13E-09	2.3 %
MCA Type3***	7.15E-08	14.4 %	1.18E-08	4.4 %
MCA Type4***	9.82E-08	19.8 %	2.90E-08	10.9 %
ALL multi compartment	1.96E-07	39.6 %	5.08E-08	19.2 %
At Power Total	4.95E-07	-	2.65E-07	-

* Not including MCA scenarios

** WRD scenarios of PAUs which did not receive DFM.

*** MCA scenario "Types" are explained in Section 25.10.2.2 (section of Task 11c).

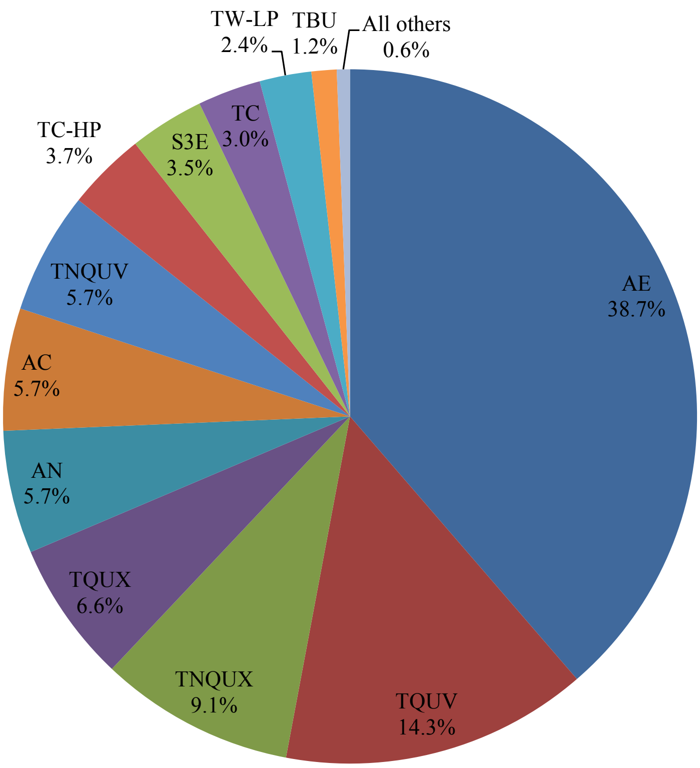


Figure 25.10.2-4 Contribution of Accident Class to Total Internal Fire at Power CDF

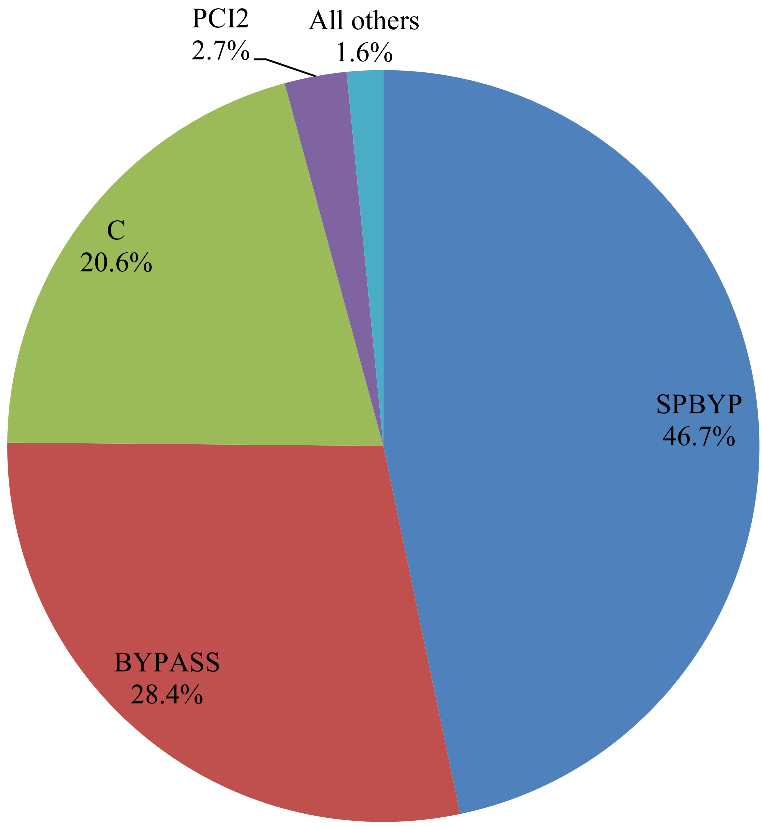


Figure 25.10.2-5 Contribution of Release Category to the Total Internal Fire LRF at Power

Table 25.10.2-7 Contributions of Release Categories to Total Internal Fire LRF at Power

Release Category	Contribution	Inclusion in LERF
S/P Bypass (ALL_VIII_except_BYPASS)	46.7 %	Y
Containment Bypass (ALL VII and BYPASS_except III)	28.4 %	Y
Early Containment Failure (ALL_IV_except_BYPASS)	20.6 %	Y
PCV Isolation Failure (PCI_II)	2.7 %	Y
RPV Rupture (ALL_IX_except BYPASS and RR and RR_LD_AE)	1.0 %	Y
Late Containment Failure (ALL_V_except_BYPASS)	0.4 %	N
Direct Containment Heating (DCH_II)	0.1 %	Y
Late Containment Failure (OP/OT_II)	0.1 %	Y
Long term SBO (Containment Failure w/o Spray _III)	0.0 %	N
Ex-vessel Fuel-Coolant Interaction (PE_II)	0.0 %	Y
Molten Core Concrete Interaction (CCI_II)	0.0 %	Y

25.10.2.4 Uncertainty and Sensitivity Analysis

The modelling uncertainties were systematically identified by reviewing the generic sources of uncertainties taken from [Ref-25.96] [Ref-25.99], and plant specific assumptions for each task. These uncertainties were

- qualitatively screened,
- qualitative assessed,
- quantitatively assessed by parametric uncertainty analysis, or
- quantitatively assessed by sensitivity analysis.

This section presents the results of parametric uncertainty analysis and sensitivity analyses. The results of uncertainty analysis and sensitivity analysis described in this section are based on the base case results presented in (1) of Section 25.10.2.3.

(1) Parametric uncertainty analysis

Parametric uncertainty analysis was performed for the Integral Fire at Power CDF and LRF using the Monte Carlo sampling method which generated a probability density function and a cumulative probability function. The parametric uncertainties of the following elements were considered.

- ignition frequencies
- fire-induced spurious event probabilities
- hot-short duration probabilities
- barrier failure probabilities
- random component failures probabilities (including CCF) / unavailabilities
- human error probabilities

The uncertainty analysis results are summarised in Table 25.10.2-8. The mean Internal Fire CDF and LRF generated based on the sample size of 100,000 are 3.76E-06 /y and 2.71E-06 /y, respectively.

Note that the point estimate values in Table 25.10.2-8 are higher than those in Table 25.10.2-1. That is because ACUBE could be applied to limited cutsets (1,000 cutsets) in the uncertainty analysis while ACUBE was applied to most of the cutsets in Section 25.10.2.3.

The parametric uncertainty analysis was not performed for the shutdown states and SFP since they are the studied by the scoping analyses in GDA.

Table 25.10.2-8 Internal Fire at Power PSA Uncertainty Analysis Results

Case	Point Estimate**	Mean	5 %	Median	95 %
CDF	2.65E-06	3.76E-06	7.65E-07	2.24E-06	9.49E-06
LRF	2.07E-06	2.71E-06	5.54E-07	1.62E-06	7.01E-06

Notes: ACUBE [Ref-25.156] is applied to top 1,000 cutsets.

Point estimates are higher than the numbers in Table 25.10.2-1 due to the limited application of ACUBE.

(2) Sensitivity Analysis

Two types of sensitivity analyses were performed for the Internal at Power Fire PSA. The sensitivity analyses, which were identified to quantitatively assess the modelling uncertainties, are presented in Table 25.10.2-9. The sensitivity analyses, which were identified to the quantitatively assess the design alternatives, are presented in Table 25.10.2-10.

The scoping analysis for the SFP (see Table 25.10.2-5) was performed under conservative approach. Simplified sensitivity analyses were performed to characterise the degree of conservatism. They are summarised in Table 25.10.2-11.

Table 25.10.2-9 Sensitivity Analyses for Internal Fire at Power PSA to Quantitatively Assess Modelling Uncertainties (1/3)

Sensitivity Analysis Case	Description	Quantitative Impact	Discussion
Case 1: Credit potential fire suppression	For all scenarios for which fire suppression have not been credited, change the non-suppression probabilities set to 0.1.	39 % reduction of total CDF 37 % reduction of total LRF	As single compartment DFM and/or multi compartment DFM were applied to limited number of PAUs at current stage, fire suppression is credited for limited number of fire scenarios. This sensitivity study shows an estimate of the refinement potential from performing additional DFM.
Case 2: Credit non-rated barriers and limit HGL propagation pathway	For all MCA scenarios that have barrier failure probabilities greater than 0.01, set the barrier failure probability to 0.01.	50 % reduction of total CDF 51 % reduction of total LRF	This sensitivity study shows an estimate of the refinement potential from specifying the non-rated boundaries that would substantially contain the damaging effects as well as from specifying the potential HGL propagation pathway over the fire area boundary.
Case 3: Credit Class 3 systems	Credit the systems which were not credited in the base case.	13 % reduction of total CDF 5 % reduction of total LRF	Screening of systems in Task 2 is justified by the small impact on especially LRF.
Case 4: Credit Class 3 systems and change default fire initiator	In addition to Case 3, change the default fire initiator (surrogate internal events) from TM (loss of condenser heat sink) to IE-TG (general transient) in order to additionally credit power conversion system.	12 % reduction of total CDF 6 % reduction of total LRF	Treatment of default fire initiator (surrogate internal events) in Task 2 is justified by the small impact from this sensitive analysis (relative to Case 3).
Case 5: Credit hot short probability for all fire scenarios	Remove the hot short events that were set to TRUE in the fire impacts table, which enables crediting the hot short probabilities derived in Task 10 for all fire scenarios including the fires starting at the PAUs containing cabinets.	6 % reduction of total CDF 7 % reduction of total LRF	Since the hot short probabilities derived from Task 10 are relatively high (up to 0.56), the impacts on CDF and LRF are not significant.

Table 25.10.2-9 Sensitivity Analyses for Internal Fire at Power PSA to Quantitatively Assess Modelling Uncertainties (2/3)

Sensitivity Analysis Case	Description	Quantitative Impact	Discussion
Case 6: Remove double-counting of fire risk	For the MCA scenarios where the exposing PAU did not receive Single Compartment Analysis (SCA) DFM, resolve double-counting of CDF and LRF among the WRD scenarios and MCA scenarios.	2 % reduction in CDF 2 % reduction in LRF	This sensitivity analysis produced a 2 % reduction in CDF and LRF, indicating very little quantitative impact for MCA scenarios where the exposing compartments did not receive detailed fire scenario selection.
Case 7: Remove fine impacts on RVI sensing lines	Remove the RVI sensing lines from the modelled cable list.	0.5 % reduction in CDF 1.2 % reduction in LRF	The reason for the small impacts on the total CDF and LRF is the small contributions of the fire scenarios involving the fire impact on the RVI sensing lines. The current treatment of the RVI sensing line in the base case is judged not to be overly conservative.
Case 8: Remove random failures of non-selected components in Task 2	Set all the basic events which are screened by the disposition codes 3 or 6 to FALSE (guaranteed success).	8 % reduction in CDF 9 % reduction in LRF	Based on these results, the completeness of the equipment selection in Task 2 is not fully demonstrated. More detailed circuit analyses will be needed once detailed circuit design is available. Regarding the reactivity control, just modelling the fire impacts on the components may in turn overestimate the risk. The fire scenarios should be carefully analysed in terms of fire growth, duration of fire and required timing of reactivity control functions.
Case 11: Update scenario frequencies	Resolve potential underestimation of plant-wide transient fire frequencies, and perform additional refinements which should have been done for the base case (if the underestimation did not occur).	5 % reduction in CDF 11 % reduction in LRF	The impact of resolving the potential underestimation of plant-wide transient fire frequencies are sufficiently cancelled out by additional refinements of Type 2 MCA scenarios which should have been performed at the base case if the frequencies were not underestimated.

Table 25.10.2-9 Sensitivity Analyses for Internal Fire at Power PSA to Quantitatively Assess Modelling Uncertainties (3/3)

Sensitivity Analysis Case	Description	Quantitative Impact	Discussion
Case 12: Remove conservatism in recovery rule	Remove the known conservatism in the recovery rule file structure.	14 % reduction in CDF 21 % reduction in LRF	This sensitivity analysis shows significant impact by removing the conservatism.
Case 13: Remove protected cables from SCA scenarios	In the SCA scenarios based on detailed fire modelling, remove fire impacts to the protected cables which are not terminated at the analysed fire compartment.	16 % reduction in CDF 17 % reduction in LRF	This sensitivity analysis shows significant impact by removing the conservatism.
Case 14: Remove long term credit to S/P given ISLOCA with potential S/P drainage	For the ISLOCAs at LPFL injection lines where S/P is potentially drained, remove long term credit to S/P as water source for HPCF.	0.04 % increase in CDF 0.05 % increase in LRF	Based on the small impact, potential S/P drainage given ISLOCA is not significant modelling uncertainty.
Case 15: Reduce hot short duration probability for spurious operations of 8 SOVs	For the cutsets where fire induced spurious operation of 8 SOVs for SRVs and only one hot short duration probability is assigned, reduce the hot short duration probability to the floor value 1.0E-05.	0.5 % reduction in CDF 0.6 % reduction in LRF	It is understood the contribution of huge number of cutsets involving 8 SOVs spurious operation were made insignificant by ACUBE and therefore the reduction of the hot short duration probability has insignificant impacts. As conclusion, the conservatism of the base case is judged acceptable.
Case 16: Update HEPs based on latest HRA report	Change the probabilities of significant HEPs in the existing CDF and LRF cutsets if base Internal Events HEPs have been updated following the base case quantification of the Fire PSA. Note that this is a simplified cutset base sensitivity analysis to understand the level of impact.	18 % increase in CDF 29 % increase in LRF	The significant increases in CDF and LRF are mostly due to the significant increases of common cognition error HEP for long term heat removal. Refinement of this HEP should be considered.

Table 25.10.2-10 Sensitivity Analyses for Internal Fire at Power PSA to Quantitatively Assess Design Alternatives

Sensitivity Analysis Case	Description	Quantitative Impact	Discussion
Case 9: Introduce rated boundary between floors in C/B	This sensitivity analysis aims at understanding the benefit if fire resistance floors are introduced in Control Building (C/B) as was assumed for Reactor Building (R/B).	5 % reduction in CDF 5 % reduction in LRF	This design alternative has small impacts on CDF and LRF because the contribution from the fire scenarios involving propagations of fire impacts through the floors in C/B is small.
Case 10: Preclude spurious events of systems in B/B	Spurious operation events for B/B fires contribute significantly to the Fire PSA risk profile. A potential plant design consideration is to preclude spurious operations by adequate circuit design.	33 % reduction in CDF 37 % reduction in LRF	Both CDF and LRF are significantly reduced, which implies the benefit of precluding spurious operations of the systems in B/B.

Table 25.10.2-11 Simplified Sensitivity Analyses for SFP Internal Fire PSA Scoping Analysis to Quantitatively Assess Conservatisms

Sensitivity Analysis Case	Description	Quantitative Impact	Discussion
Consider realistic time available for terminating fire-induced flooding	The time available to manually trip FLSS pumps was assumed to be 1 hour based on the design flow rate. In the sensitivity analysis, the time available is changed to 4 hours based on the expected realistic flow rate of SFP overfill.	15 % reduction in FDF	This sensitivity analysis shows significant impact by removing the conservatism.
Consider realistic consequence from PCV failure without core damage (Level 1 PSA success sequences)	PCV failure without core damage (Level 1 PSA success paths with containment failure) is assumed to fail all the systems in R/B with a 1.0 conditional probability due to the potential steaming inside the R/B. In the sensitivity analysis, the conditional probability was reduced to 0.1 by considering that the drywell head flange is likely to fail first given PCV overpressure.	20 % reduction in FDF	This sensitivity analysis shows significant impact by removing the conservatism.
Remove FDF which is double-counting the LRF of reactor	Approximately 30 percent of the FDF for SFP POS F involves loss of SFP cooling or SFP leakage induced by containment failure after core damage (large release sequences in at Power Fire PSA). This fraction of FDF is double-counting the at Power Fire LRF.	30 % reduction in FDF	This sensitivity analysis shows significant impact by removing the conservatism.

25.10.2.5 Insights from Assessment

From the results, significant Internal Fire risk insights can be developed.

(1) At Power

- Fires originating in B/B contribute the greatest fire risk of any plant area. This is in large part due to the potential plant impact due to spurious operation, such as Safety Relief Valve (SRV) opening or containment vent valve opening/isolating.

- Type 2 MCA scenarios (local effects on adjacent compartments through un-sealed penetrations including non-rated doors) have large contributions to the total calculated Internal Fire CDF/LRF. This is in large part due to the fire impacts to both divisions of Class 2 systems in B/B.

- Loss of the systems located at / controlled from B/B, combined with Class 1 software failures, is the dominant accident sequence, which is consistent with the Internal Events at Power PSA.

- The availability of 2 Remote Shutdown System and an independent control room in the B/B significantly reduces the contribution of MCR fire scenarios.

- Fire risk originating in the Reactor Building (R/B), Control Building (C/B) and Heat Exchanger Building (Hx/B) is relatively low, which implies the adequacy of the divisional segregation.

- The cutset reviews performed during the model setup and debugging indicated that the potential for spurious opening of SRVs due to a hot short is one of the main reasons for challenging the plant response to a fire event. This observation was confirmed by the calculated CDF/LRF results.

Note: The above insights are superseded by those in (4) at Power (Refinement).

(2) Shutdown (scoping analysis)

Based on this scoping analysis, it was concluded that FDFs during POS B-2 and POS C were sufficiently lower than the Internal Fire at Power LRF. Since POS B-2 and C are deemed dominant contributor to the FDF due to fire during Shutdown, the fire risk during Shutdown is expected to be also insignificant compared to the at power fire risk. The specific insights are listed below.

- Fires in PCV (unique to shutdown states due to de-inerted condition) have small contribution to the fire risk during shutdown. This is because only the spurious signal from the shutdown cooling inboard isolation valve (for the operating RHR train) could cause a fire-induced initiating event.

- The large contribution of B/B fires comes from the potential plant impact due to spurious injection to RPV or SFP by FLSS which is assumed to cause induced flooding.

- The large contribution of Turbine Building (T/B) fires comes from the potential plant impact due to loss of offsite power, as well as the high transient fire frequencies in T/B during shutdown.

(3) SFP (scoping analysis)

Based on this scoping analysis, the Fire FDF of SFP at Power is approximately 37 percent of the Internal Fire at Power LRF for reactor. This is due to a number of conservatisms in place in the current scoping analysis. By considering the sensitivity analyses in Table 25.10.2-11, the FDF would be much smaller. Factoring the long time available (more than 300 hours) would further reduce the FDF. It is concluded that the FDF of SFP during POS F is also insignificant compared to the at Power fire risk of reactor.

(4) At Power (Refinement)

- Fires originating in the R/B electrical rooms Divisions I and II contribute the most of any plant area in terms of CDF and LRF. This is in large part due to existence of a large number of ignition sources (cabinets) and critical cables. Significant reductions of CDF and LRF from these compartments are expected once detailed cable tray information (e.g., raceway type, distance to sources) is available.

- Fire risk originating in R/B is high (nearly 70 percent of total LRF). This is in large part due to the high contributions from the R/B electrical rooms Divisions I and II as well as the Type 4 MCA scenarios impacting two divisions of RVI transmitters.

- Fire risk originating in B/B is relatively low, which implies fire rated boundaries among B/B floors and B/B divisions for defence-in-depth benefit.

- Fire risk originating in C/B is even lower, which implies fire rated boundaries among C/B floors and within C/B divisions for defence-in-depth provides significant benefit.

- MCA scenarios have small contributions to total CDF/LRF compared to SCA scenarios. This is in large part due to adoption of fire rated boundaries within each fire area for defence-in-depth.

- The availability of 2 Remote Shutdown System and an independent control room in the B/B significantly reduces the contribution of MCR fire scenarios.

25.10.2.6 Key Assumptions and Study Limitations

Assumptions in the Internal Fire PSA were made in the development phase. They relate to each Tasks define in Section 25.10.2.2. In addition, when the level of plant requirements specification limits the detail of the PSA, then appropriate assumptions were developed to allow completion of the PSA.

Starting from assumptions, sensitivity analyses have been performed. Among assumptions in the Internal Fire PSA, key assumptions, which have comparatively large impact on the result, have been listed from the result of sensitivity analyses.

Task 1: Plant Boundary and Partitioning:

- All the PAU boundaries within a fire area are conservatively assumed to be non-rated barriers in the base case. For the refinement of Internal Fire at Power PSA, this assumption is refined based on newly available design information where a number of intra-divisional boundaries are specified as 1, 2 or 3 hour fire rated. Fire rated boundaries within a fire area are credited in the refinement analysis.

- Non-rated barrier elements are conservatively assumed to not substantially contain the damaging effects of fire. This approach is conservative, and significantly impacts the Fire PSA results as demonstrated by the Sensitivity Analysis Case 2 in Table 25.10.2-9.

Task 2: Component Selection

- Since cable routing information for the balance of plant (BOP) and their support systems is not available and burdensome to develop, it is assumed that these systems are unavailable for accident mitigation. This approach is conservative, and significantly impacts the Fire PSA results as demonstrated by the Sensitivity Analysis Cases 3 and 4 in Table 25.10.2-9.

Task 3 and 9: Cable Selection and Routing

- The usual practice for existing plants is to associate cables with specific cable tray and conduit raceways. However, since the detailed cable design is not available, cables are associated with designated plant rooms through which they are assumed to be routed.

Task 11a: Single Compartment Analysis

- Fire suppression was credited for limited number of PAUs for which SCA DFM was performed. This approach is conservative, and significantly impacts the Fire PSA results as demonstrated by the Sensitivity Analysis Case 1 in Table 25.10.2-9.

Task 11c: Multi-Compartment Analysis (MCA)

- Doors and penetrations that are not listed as being 3 hour fire resistant are assumed to be non-fire rated and are treated as an unsealed opening in the base case. This approach is conservative, and significantly impacts the Fire PSA results as demonstrated by the Sensitivity Analysis Case 2 in Table 25.10.2-9. For the refinement of Internal Fire at Power PSA, this assumption is refined based on newly available design information where a number of intra-divisional boundaries are specified as 1, 2 or 3 hour rated. A 1 or 2 hour fire resistant is also credited in addition to 3 hour fire resistant in the refinement analysis.

- Fire suppression was credited for limited number of PAUs for which MCA DFM was performed or individual refinements were performed. This approach is conservative, and significantly impacts the Fire PSA results as demonstrated by the Sensitivity Analysis Case 1 in Table 25.10.2-9.

Shutdown and SFP Scoping Analysis:

- During POS B-2 and C where Class 1 AC power Division I and III are assumed in maintenance, the components powered by Class 1 AC power Division I or III, e.g., motor-driven pumps (MOPs) and motor operated valves (MOVs), modelled in the Shutdown PSA are assumed to be de-powered and thus fire-induced spurious operation does not occur. This is a simplified assumption for the scoping analysis in GDA. However, depending on the future licensee, some components (e.g., MOVs not connected to the primary coolant boundary or forming divisional boundary) may be kept in service by cross-tie of AC power. Therefore, this assumption should be re-visited in the future analysis (post GDA phase) once the maintenance practice is established.

- If FLSS pump(s) is spuriously started and injection valve(s) to SFP is spuriously opened, SFP overfill occurs during Shutdown POS B-2, C and SFP POS F. If the pumps are not manually terminated an assumed duration, the spill-out water impacts all the pumps in the R/B. Injection to RPV and SFP by MUWC is also impacted due to flood submergence of RHR MOVs.

- Local operations inside a building credited in the internal events Shutdown and SFP PSAs, e.g., alignment of manual valves for injection by MUWC or Fire Protection system, are also credited in the scoping analysis.

25.10.3 Internal Flooding PSA

This section of the Summary Report presents a summary of the Internal Flooding PSA [Ref-25.101].

25.10.3.1 Scope

The scope of work for the Internal Flood PSA (IFPSA) for GDA was as follows:

- The analysis examined the potential risk from internal flooding events originating anywhere within the site boundary.
- Internal flooding events that arise due to a breach of plant fluid systems or tanks caused by equipment failure or human error were considered. The effects of damage resulting from immersion, spray and high-energy line breaks (HELBs) were addressed.
- Accidents related to the potential release of radioactive material from the reactor vessel and containment was addressed within the IFPSA. In order to achieve this, it was necessary to propagate the internal flood analysis through both the Level 1 and the Level 2 analyses including any flood-related impacts which affect the containment integrity directly or systems credited for ensuring the protection of its integrity.
- Core Damage Frequency (CDF) and Large Release Frequency (LRF) were quantified for the at Power state (including full power and low power).
- Fuel Damage Frequency (FDF) in the reactor during shutdown was quantified as a scoping analysis for representative Plant Operating States (POS) to demonstrate that the risk from internal flood during shutdown is insignificant.
- Release from the SFP was also considered. FDF at Power (representative POS) was quantified as a scoping analysis to demonstrate that the risk from internal flood on the SFP is insignificant.
- Releases from the radioactive waste management system were excluded from the IFPSA.
- Fresh fuel in the fuel route was evaluated qualitatively to demonstrate that sub-criticality is maintained due to physical separation.
- Spent fuel in the fuel route was evaluated qualitatively to demonstrate that sub-criticality is maintained due to physical separation and the consequences of radioactive release are small using passive protection mechanisms.

25.10.3.2 Methodology of Tasks for the Internal Flooding PSA

The IFPSA was performed based on the guidance provided by EPRI [Ref-25.102]. The full EPRI methodology [Ref-25.102] and ASME/ANS PSA Standard - RA-Sb-2013 [Ref-25.13] were reviewed.

The IFPSA method is comprised of ten technical tasks and a documentation task. Detailed task plans for each of the ten technical tasks were produced based on a process designed to meet the relevant requirements of the combined ASME/ANS standard. Some customisation of the standard approach was necessary given the “yet to be constructed” status of the plant, but these adjustments specific to each task are referred to in the sub-sections that follow.

Task 1: Define Flood Areas

In Task 1, the plant buildings and rooms were reviewed in order to determine the scope of the IFPSA, define flood areas, and determine the associated inter-area flood propagation pathways.

The first step was to define the global plant area boundary with respect to internal flooding, which was defined in a similar manner to that of the Fire PSA Task 1.

The second step was to define appropriate flood areas with three considerations in mind:

- First, flood areas facilitated the identification of areas where a flood may impact systems, structures and components (SSCs) that could cause an initiating event or need for immediate plant shutdown, damage equipment needed to respond to an initiating event, or both.

- Second, flood areas facilitated the identification of sources of flooding as well as flood failure mechanisms that needed to be considered.

- Third, flood areas were defined to characterise the different flood propagation pathways that needed to be considered. Plant mitigating features (e.g., floor drains, sumps, sump pumps) were identified within each flood area, as well as plant penetrations that can serve to propagate a flood from one area to another (doors, HVAC ducts, louvers, windows, etc.).

It should be noted that the identification of SSCs and flood sources is formally part of Task 2 and the specific PSA-related SSCs and flood sources were documented within Task 2; however, a general indication of whether or not buildings contain PSA-related SSCs or flood sources was performed in Task 1. This is because the identification of SSCs and flood sources was an important part of conducting the preliminary screening exercise based on buildings/structures, and this exercise reduced the overall effort associated with the IFPSA.

All the flood area definitions for the at Power analysis were retained for the scoping analysis of Shutdown POSs. Degraded boundaries due to maintenance periods were considered in Task 7 (Flood Consequence Analysis).

Task 2: Identify Flood Sources, Flood Mechanisms, and SSCs

The purpose of this task was to identify the potential flood sources, their associated flood mechanisms, and the SSCs that can be impacted by the various flood sources. Essentially, this task involved the systematic identification and characterisation of the flood sources and flood mechanisms associated with each flood area identified in Task 1. For the most part, the SSCs were already identified in Task 1, but they were documented in Task 2.

In Task 2, the potential flood sources (liquid/steam plant systems, tanks, etc.) were identified and characterised. In addition to flood source characteristics such as pressure, volume, flow rate, etc., specific flood mechanisms were identified for each flood source including failures of pipes and associated fittings, spurious actuations, and human-induced floods.

It should be noted that the information available and relevant to the internal flood source identification task was, for the most part, plant-specific.

Task 3: Conduct Plant Walkdowns

The purpose of Task 3 was to verify the information collected in Tasks 1 and 2 and to augment the data collected in Tasks 1 and 2 based on physical observations (e.g., potential for spray, special insights, etc.).

Physical plant walkdowns of the plant were not practical as the plant is still in the design phase. Instead, a virtual plant walkdown was performed using the 3D CAD model.

Task 4: Qualitative Screening of Flood Areas

In Task 4, a screening evaluation of all areas of the plant was performed based on criteria that consider three aspects of flood area importance: flood sources, flood propagation pathways, and flood consequences in terms of initiating event and the impact on SSCs needed to prevent core damage and large release. In the screening process, each flood area was reviewed and either retained or screened out.

Task 5: Characterise Flood Scenarios

The following sub-sections outline the process of the initial scenario development.

Plant Familiarisation

The various plant structures identified in Tasks 1-4 were reviewed for their physical locations and inter-building connections.

For each identified plant structure, a detailed review of the structure was performed to learn the general characteristics of the structure, its contents, the interfaces with other structures, the mitigating features of the structure, and to develop a general perspective for how floods would propagate through the structure. This was accomplished by review of the flood area plant layout drawings and the information collected and documented in Tasks 1-4. This was documented in a short narrative for each building and structure.

Scenario Definition

The basic elements of a scenario definition were considered and included the following:

- POS
- Flood Area
- Flood Source
- Flood Type and SSC Susceptibility
- Propagation Pathways

When a structure was selected and the familiarisation process was completed, scenario development was performed. Each defined flood area containing flood sources as identified in Tasks 1-4 was evaluated, starting with the top floor and working progressively down though the structure until all flood areas with flood sources were considered.

Due to consideration of possible combinations of flood source, water spray impact, flood size, propagation path, and equipment submergence, a large number of flood scenarios were involved. The representative flood scenarios were identified by plant location rather than postulating all possible scenarios for each possible source in order to develop a more manageable number of defined scenarios.

Task 6: Flood Initiating Events Analysis

The frequency of flooding due to component failure was determined based on industry leakage and rupture data. For each credible flood source in a flood area, all the potential flood inducing components were considered in the derivation of the flood initiating event frequency. The components included valves, expansion joints, pipes, heat exchangers, tanks, etc.

The source of the generic data for pressure boundary failures that was used for this task was the recent EPRI technical report [Ref-25.42]. The piping system failure rates in the EPRI report are presented in three different pressure boundary failure modes (i.e., spray, flood, and major flood) for nine different systems accounting for U.S. service experience from 1970 through 2012. This data source has accounted for the degradation and damage mechanisms that are unique to these system groups. These system-specific pipe failure rates and rupture frequencies are represented in terms of failures per linear foot and reactor operating year (by pipe size group). They account for leakage and rupture of piping components in these systems (implicitly including pipes, valves, pumps, heat exchangers, tanks, fittings, flanges, gaskets, etc.).

In the EPRI report, piping system rupture frequencies are calculated as the product of the system-specific pressure boundary failure rates and the conditional probability of pressure boundary rupture in a specified break size range. This is also calculated separately by pipe size group. The conditional rupture size probability is determined in the EPRI report from service experience, insights from reviewing service data, and engineering judgment, with uncertainty treated using the Beta Distribution.

Plant-specific information on pipe diameter and pipe length was collected for each flood area identified in Task 2. This information was then used in conjunction with the EPRI leakage/rupture rates to calculate the plant-specific pressure boundary failure rates and the corresponding initiating event frequencies.

As noted above, these failure rates implicitly include valves, pumps, heat exchangers, tanks, fittings, flanges, gaskets, etc. in addition to piping. As such, individual component failure frequencies are not separately developed since they are by definition included in the initiating event frequency calculations.

For each flooding scenario defined in Task 5, the initiating frequency was estimated based on the piping in the area in which the scenario is initiated. The initial quantitative evaluations used the total submergence frequency and the spray frequency as appropriate for the initial scenarios defined in Task 5.

It should be noted that no quantitative screening of the internal flood scenarios was performed based on the calculated flooding initiating event frequencies. The only screening performed was as part of Task 4 and this was performed on a qualitative basis only.

Task 7: Flood Consequence Analysis

The initial flood scenarios were developed and evaluated in Tasks 5 and 6. In some cases, these initial scenarios were of a low enough frequency that there is limited benefit in refining them. However, many of the initial scenarios resulted in a higher CDF or LRF and warranted refinement or, in some cases, detailed analyses.

Screening of Flood Scenarios

The ASME/ANS standard supporting requirement IFQU-A3 for Capability Category II permits quantitative screening of scenarios where the product of the IE frequency and the conditional core damage probability (CCDP) is less than 10-8/y. However, because the risk associated with the UK ABWR is significantly lower than an existing BWR (for which the ASME/ANS standard criteria were set); no screening of flood scenarios was conducted. All developed flood scenarios were either retained at their initial values or replaced with refined scenarios.

Refined and Detailed Analysis

The basic characteristics of each flood scenario were developed in Task 5. These characteristics were reviewed to identify conservatisms in the scenario modelling to be reviewed for refinements or detailed analysis. The following sections describe some of the options available to refine flood scenarios.

Flood Sources and Volumes

In the initial analyses, three types of flood scenarios were developed: spray, submergence, and HELB (assuming the line is high energy). The initial scenarios were developed assuming that the flood volume is infinite. One option used to refine flood scenarios was to base the scenario on the actual system volume and flood timing.

If a scenario had multiple systems contributing to the flood initiating event, various refinements were made by sub-dividing the scenario into separate scenarios, e.g., based on system flow rates.

Flood Initiating Event Types

In the initial analysis, a unit trip or a manual shutdown was assumed for all identified flood areas containing a flood source. When multiple source systems are present in a scenario the most onerous initiator type was used. In cases where the flood source is a standby system or train, the pipe failure may not have a direct impact on the operating plant.

If multiple flood sources are present in a scenario, they may result in different initiating event types. In the initial scenarios, the most conservative of the initiating event types was used. However, the scenario can be separated into separate scenarios for each contributing flood source and the appropriate initiating event type used.

In cases where the flood source is a standby system or train, the pipe failure may not have a direct impact on the operating plant. In these cases, the plant Technical Specifications were used to determine if the event would require a manual shutdown.

Flood Height Calculation

In the initial flood scenarios, it was assumed that all SSCs were failed (as appropriate for the flood type) in the flood source room and in every room to which the flood propagates. This is a conservative assumption. By determining the actual flood depth(s), the set of impacted SSCs for a scenario was reduced.

To characterise the consequences for each flood-induced initiating event, the type of flood source, spill rate, flood location, area discharge rate, flood accumulation rate, flood height, time to reaching a critical flood depth, the capacity of drains, the amount of water retained by sumps, berms, dykes and kerbs, and the impact on accident initiation/ mitigation equipment was considered.

The flood depth was determined by the product of the flood accumulation rate (i.e., the spill rate minus the drainage or discharge rate) and the time to deplete the available flooding inventory, divided by the unoccupied floor area. No flood height calculation was needed in areas in which there are no flooding sources in the area or in its adjacent areas, no barriers exist in a large open area to contain the floodwater in the area, and components in that area cannot be damaged from water drip or spray.

The water discharge from an area may be via gaps beneath the doors (doorways), drains (e.g., floor drains), sump pumps, and any other potential pathways (e.g., openings in the equipment hatches).

Floor Drains

It should be noted that, in the current analysis, flood drainage was not credited. However, the floor drains will be credited in future analysis at which the approach presented here will be used.

One of the following two factors will be considered to determine the drainage capacity of the floor drains:

- The capacity for flow through drain header piping
- The capacity for flow passing through the floor drainage screens

Flow rates through the drains can be estimated by determining the difference in height between the level of water in the room and the level where the drain header opens up into larger piping. If the floor drains are arranged so that the drain header capacity is not reached, then drainage rates will be determined by calculating the allowable flow through the drainage screens at the mouth of each drain. The maximum drain flow rate is usually estimated using a full pipe flow model that assumes the head over the drain is sufficient to fill the drain body completely. At low flooding rates, drain flow would be somewhat less. For these scenarios, the condition resembles that of flow through a broad crested weir and the discharge rate can be estimated by the formula developed for the broad crested weir.

Flow under doors (gap) can be modelled by flow through a submerged sluice gate structure. This also conservatively assumes the existence of a large volume flooding source (i.e., tank, large pipe) capable of filling the room with water shortly after the rupture. The steady state flow rate into the room is assumed to be the incoming flood rate, minus any drainage paths other than those under the door. It also assumes that the downstream flow region is wide and the critical flow region just downstream of the door does not flood, which is reasonable unless a kerb exists outside the door.

Other than equipment damage due to spraying and dripping, the critical flood volume necessary to damage accident initiation/mitigation equipment is estimated based on the estimated area/volume of the rooms in question, the fraction of empty space in the room, capacity of drains, water retention by curbs/dikes/sumps/berms, and the depth at which the accident initiation/mitigation equipment would be damaged. The flood damage timing, if needed (i.e. time to reach critical flood height), will be determined by dividing the critical flood volume by the flood accumulation rate. The flood accumulation rate is determined by subtracting the area discharge rate (based on the pathways out of the compartment) from the flow rates into the compartment (based on the flood spill rate and/or the pathways into the compartment). The above information is used in conjunction with susceptibility of equipment to flooding damage to make the final determination as to the set of accident initiation/mitigation equipment impacted.

Evaluating Equipment Damage

Given that a susceptible component is flooded, credit may still be taken for the availability of the equipment with respect to potential flooding effects if supported by an engineering analysis (e.g., environmental qualification analyses), test or operational data, or expert judgment.

Task 8: Human Reliability Analysis

Task 8 was performed in the following steps:

Step 1: Operator actions were identified and the human failure events (HFEs) to be included in the IFPSA model were defined

Step 2: Human error probabilities (HEPs) were quantified

For risk-significant HFEs identified by the IFPSA, a detailed HRA analysis was performed following a similar approach to that performed for the Internal Events HRA assessment. For non-risk-significant HFEs, a screening approach was followed.

Step 3: HRA dependency analysis was carried out

Step 4: Uncertainty and sensitivity analysis was performed

Task 9: PSA Modelling of Internal Flood Scenarios

Task 9 is associated with undertaking the modelling of internal flood scenarios within the existing PSA and focused on ensuring that the flood scenarios that were taken forward for quantification had the appropriate accident sequences assigned to them, and that impacts on systems and operator actions from flooding were captured correctly.

To quantify flood-induced accident sequences, the flood scenarios started with the base Internal Events PSA models (at Power, shutdown and SFP). This task thus involved the process of modifying the existing Internal Events PSA model to include the required internal flooding initiators and the linkage of flooding equipment failures (including direct effects such as loss of a ruptured system and indirect effects such as submergence, jet impingement, and pipe whip; as applicable), human actions, and recoveries.

Based on the flood-induced failures and actions required in the plant procedures, appropriate accident initiators and sequences were selected to model the flood-induced accident sequences in Tasks 5 and 7. The general transient event tree structure was used for the majority of the internal flooding scenarios that do not result in the loss of the flood source system in order to assess the potential accident sequences derived from the respective flooding initiator. Other event tree structures, such as loss of a cooling water system, were used when that system is the source for a specific flood initiator.

The accident sequences for the selected event trees were reviewed to confirm the applicability of the model to the flooding scenarios. Where the existing event tree(s) did not appropriately reflect all of the accident sequences associated with the flooding scenario, the sequence logic was modified as necessary to account for any unique flood-induced scenarios and/or phenomena. In addition, the effects of selected equipment failures, unique operator actions that are not included in the existing Internal Events PSA model, and any other necessary changes or additions to the existing Internal Events PSA model were added to accurately reflect the plant response and mitigation actions. The model modifications included incorporation of recovery actions to mitigate the effects of flood-induced loss of components.

Internal flooding can affect the release frequency by changing the frequencies of sequences leading to release, or by introducing new sequences leading to release.

The flooding condition may introduce a new failure mode, cause the loss of a containment function, or preclude the possibility of an operator mitigation action, depending on the location of the break that leads to the flooding and the extent of the effects of the flooding event. All of these considerations have been accounted for in the review and revision of the accident sequence and system models to develop the integrated PSA model for the flooding scenarios.

Task 10: PSA Quantification

Task 10 is associated with the quantification of the flooding scenarios after integration into the existing Internal Events PSA model. Iterations and revisions were required once the initial quantification had taken place to address overly dominant contributors to risk and to refine over conservative modelling.

Such revisions were related to:

- Accident sequences - for each flooding scenario, the associated accident sequences were reviewed for applicability. Where appropriate sequences did not exist then modifications to existing accident sequences were performed to account for any unique flood-induced accident sequences and associated phenomena.

- System Analysis - the system modelling was amended to account for the impacts from flooding on the operation of that system. This was accomplished through the insertion of a flooding initiating event or house event logic.

- Data Analysis - any additional analysis of SSC data required to support quantification of flooding scenarios was done such that it was consistent with existing data analysis procedures.

- Human Reliability Modelling - any revision to the modelling of human actions to account for the specific flooding scenarios noting changes in response times, recovery mechanisms, stress levels etc. was accomplished through the use of screening values or detailed HRA assessments.

- Sequence Quantification - sequence quantification was carried out to calculate the risk contribution of the flooding scenarios. The issue of combined effects of failures due to flooding and those coincident with the flooding due to independent causes have been accounted for. Similarly, the direct and indirect effects of failure from flooding have been accounted for, e.g. loss of service water system due to pipe rupture (direct) coupled with loss of electrical system due to spray effects (indirect).

25.10.3.3 Quantification Results

(1) At Power results

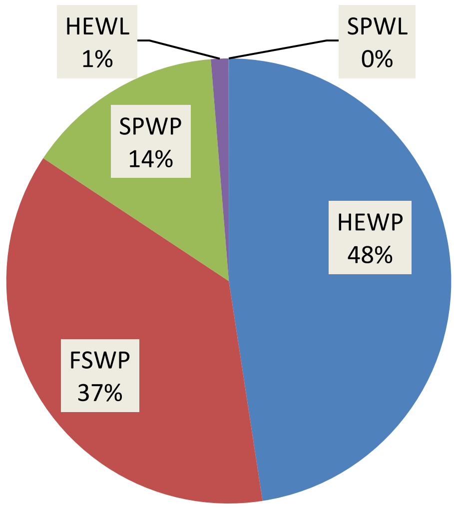
The Internal Flooding base case CDF is 1.76E-06 /y and the Internal Flooding LRF is 7.81E 07 /y [Ref-25.101]. There are some known conservatisms that were explored in the sensitivity analysis. The nature and extent of further information that is required for refinement are listed in Section 25.10.3.6.

The Internal Flooding LRF is broken-down by the scenario types in Figure 25.10.3-1. The highest contributor is the high energy line breaks and pipe whip (HEWP) scenarios followed by the flood submergence with propagation (FSWP) scenarios and spray with propagation (SPWP) scenarios.

The Internal Flooding LRF is broken-down by the buildings in Figure 25.10.3-2. The LRF is dominated by flooding in the Reactor Building (R/B).

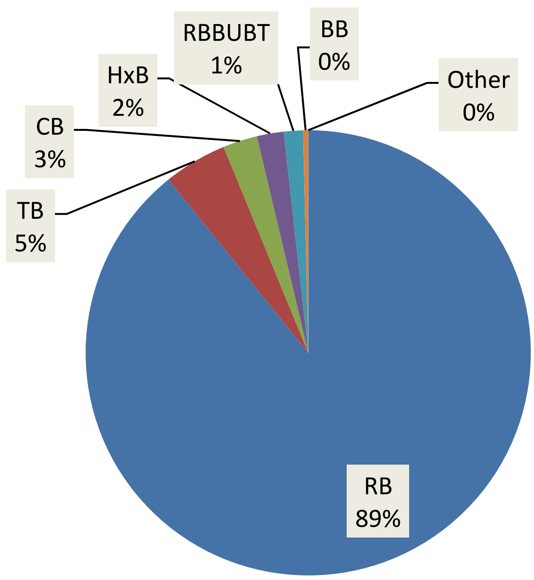
Scenarios that are relatively more important for the Level 2 PSA are those that result in anticipated transients without scram (ATWS) due to flood impacts on the reactor protection system (RPS) signal. For the ATWS PDSs, there is no Level 2 mitigation applied to the Level 1 cutset. Other scenarios that are more important in the Level 2 PSA results than the Level 1 PSA results include break outside containment

(BOC) scenarios with failure of containment isolation, which means there is no Level 2 mitigation applied to the Level 1 cutset. Similarly, scenarios which drain down the suppression pool due to a pipe break are relatively more important in the Level 2 PSA, as the pipe break fails containment isolation.



HEWP: High Energy Line Break and Pipe Whip with propagation
FSWP: Flood Submergence
SPWP: Spray with Propagation
HEWL: High Energy Line Break and Pipe Whip without propagation
SPWL: Spray without Propagation

Figure 25.10.3-1 Break-down of Internal Flooding at Power LRF by Scenario Types



RB: Reactor Building
TB: Turbine Building
CB: Turbine Building
HxB: Heat Exchanger Building
RBBUBT: R/B-B/B Connecting Service Tunnel
BB: Backup Building

Figure 25.10.3-2 Break-down of Internal Flooding at Power LRF by Buildings

(2) Scoping analysis results for shutdown states and SFP

In the scoping analysis, quantification was limited to POS B-2 and POS C. POS C was selected due to the highest risk contribution in the Internal Events Shutdown PSA (see Section 25.8). POS B-2 was included in the unique condition, i.e., SFP is covered due to open pool gate, and less mitigation systems available compared to POS B-1.

The total calculated Internal Flooding FDFs at Shutdown POS B-2 and POS C are 1.20E-09 /y and 3.44E-08 /y, respectively [Ref-25.101].

In the scoping analysis, quantification was limited to POS F (at Power). POS F for SFP PSA was selected due to the longest duration (highest ignition frequency) and also because risk of SFP during shutdown was studied in the scoping analysis of Shutdown POS B-2.

The total calculated Internal Flooding FDF of SFP at Power (POS F) is 1.33E-07 /y [Ref-25.101].

(3) At Power refinement results

The significant conservatism in the Internal Flooding at Power PSA base case, described in (1) of this section, made the calculated overall risk of UK ABWR generic design conservatively high and impacted the use of PSA to inform the design process for ALARP demonstration.

The Internal Flooding at Power PSA was significantly refined from the base case. The purpose of the refinement is to calculate a more realistic flood risk of UK ABWR in GDA phase, and to make the UK ABWR PSA more useful for the ALARP demonstration.

As the results of various refinements of risk significant scenarios, the Internal Flooding refinement CDF is 1.75E-06 /y and the Internal Flooding LRF is 1.75E 07 /y. The LRF by flood at Power reduced from 7.81E-07 /y to 1.75E-07 /y.

The Internal Flooding LRF is broken-down by the scenario types in Figure 10.3-3. The highest contributor is the spray with propagation (SPWP) scenarios followed by the flood submergence with propagation (FSWP) scenarios and high energy line break and pipe whip (HEWP) internal blast scenarios.

The Internal Flooding LRF is broken-down by the buildings in Figure 10.3-4. The LRF is dominated by flooding in the Reactor Building (R/B).

The most significant reduction of LRF was achieved for High Energy Line Break scenarios, since a number of significant scenarios were refined. The flood submergence scenarios were also benefitted mainly from new credit of manual SCRAM and general refinement.

The reduction of LRF was significant while CDF reduced only little. That is because the scenarios were refined from the top contributors to LRF. That is, the scenarios with higher LRF to CDF ratio (up to 1.0) were prioritised for the refinements. Another reason is removal of credit to manual isolation within 1 hour which impacts CDF more significant than LRF (see key assumption in 25.10.3.6).

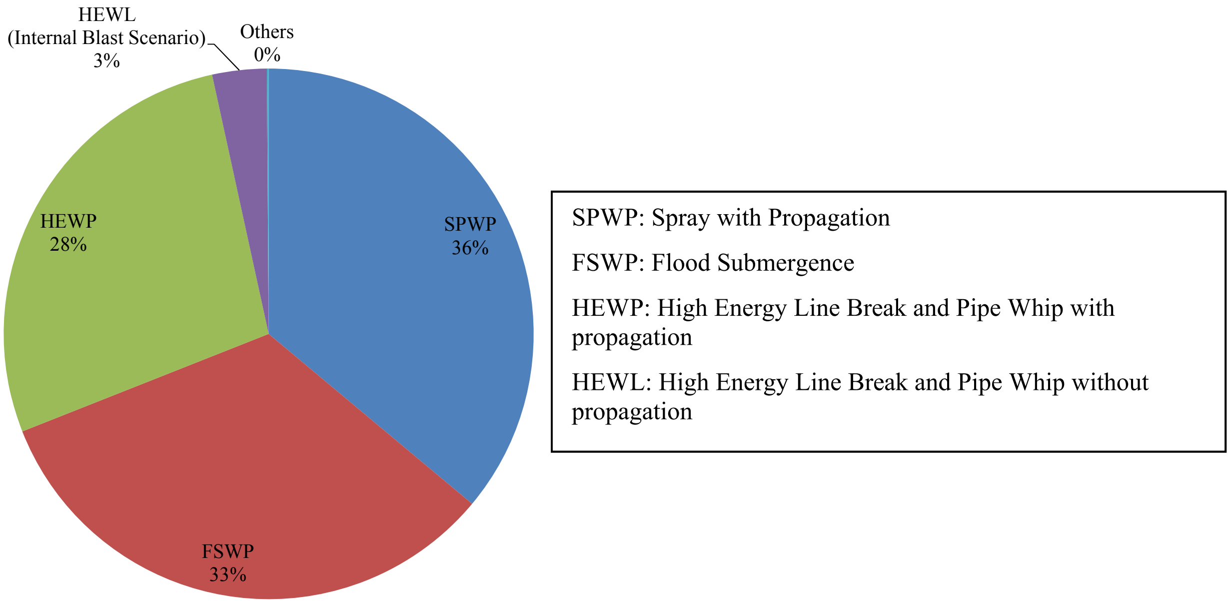


Figure 25.10.3-3 Break-down of Refined Internal Flooding at Power LRF by Scenario Types

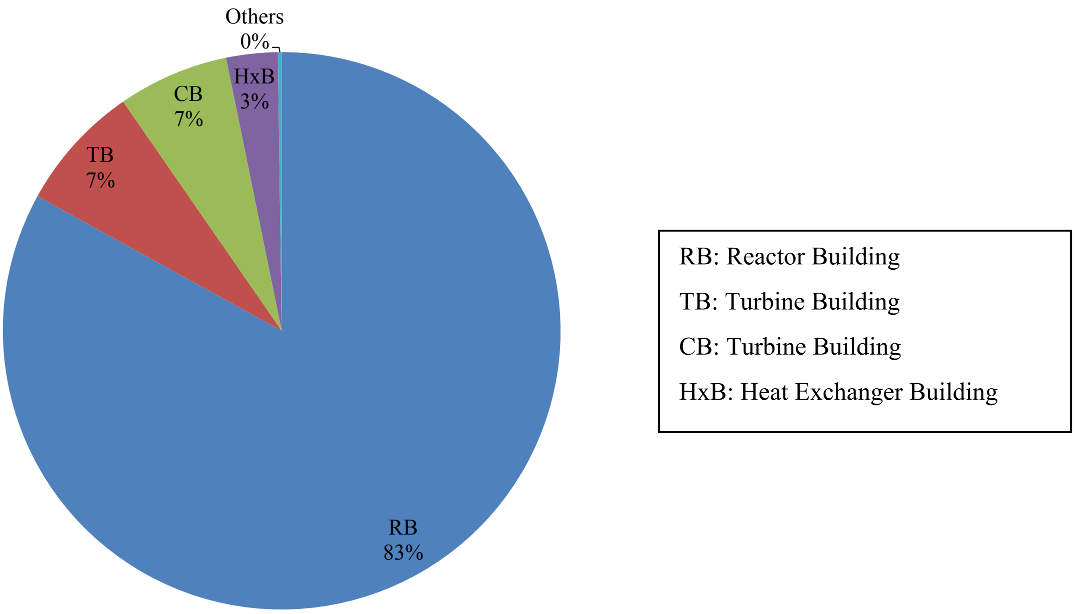


Figure 25.10.3-4 Break-down of Internal Flooding at Power LRF by Buildings

25.10.3.4 Uncertainty and Sensitivity Analysis

This section presents the results of the parametric uncertainty analysis and sensitivity analyses [Ref-25.101]. The results of uncertainty analysis and sensitivity analysis described in this section are based on the base case results presented in (1) of Section 25.10.3.3.

(1) Parametric uncertainty analysis

Parametric uncertainty analysis was performed for the Internal Flooding at Power CDF and LRF using the Monte Carlo sampling method which generated a probability density function and a cumulative probability function. The parametric uncertainties of the following elements were considered.

- Pipe failure frequencies
- Random component failures probabilities (including CCF) / unavailabilities
- Human error probabilities

The uncertainty analysis results are summarised in Table 25.10.3-1. The mean Internal Flooding CDF and LRF generated based on the sample size of 10,000 are 2.31E-06 /y and 1.03E-06 /y, respectively.

The parametric uncertainty analysis was not performed for the shutdown states and SFP since they are the studied by the scoping analyses in GDA.

Table 25.10.3-1 Internal Flooding at Power PSA Uncertainty Analysis Results

Case	Point Estimate	Mean	5 %	Median	95 %
CDF	1.76E-06	2.31E-06	6.30E-07	1.50E-06	4.81E-06
LRF	7.80E-07	1.03E-06	2.72E-07	6.49E-07	2.24E-06

(2) Sensitivity Analysis

Two types of sensitivity analyses were performed for the at Power IFPSA. The sensitivity analyses, which were identified to quantitatively assess the modelling uncertainties, are presented in Table 25.10.3-2. The sensitivity analyses, which were identified to quantitatively assess the design alternatives, are presented in Table 25.10.3-3.

The scoping analysis for the SFP was performed under a conservative approach. Simplified sensitivity analyses were performed to characterise the degree of conservatism. They are summarised in Table 25.10.3-4.

Table 25.10.3-2 Sensitivity Analyses for Internal Flooding at Power PSA to Quantitatively Assess Modelling Uncertainties (1/2)

Sensitivity Analysis Case	Description	Quantitative Impact	Discussion
Case 1: Credit components without cable routing in HEWP scenarios	The components with electrical dependency and unknown cable routing are assumed to always fail in HEWP scenarios. In the sensitivity analysis, they are assumed not failed.	8 % reduction of total CDF 3 % increase of total LRF	The small increase in the LRF is due to the change of PDS from those with intact containment to those with failed or bypassed containment.
Case 2: Pipe whip / jet impact on check valves in Main Steam tunnel	The check valves in the Main Steam System (MS) tunnel were assumed to fail given a break of any high energy line in the MS tunnel. In the sensitivity analysis, they are assumed not failed if the diameter of broken pipe is less than 6 inches.	98 % reduction in CDF of the HEWP scenario (contributing 14 % to total CDF in base case).	It is important to assess the realistic capability of check valves to withstand against pipe whip / jet impact given a break of small diameter pipe.
Case 3: Impact of high energy line break at turbine hall	It was assumed for the refined high energy line break scenario in the turbine hall that high energy steam is released from the blowout panel, such that offsite power was available. A sensitivity analysis is performed to understand the impact if steam propagates to the flood area containing equipment for keeping offsite power and thus loss of offsite power.	8 % increase in total CDF 6 % increase in total LRF	The assumption in the base case is deemed realistic because there is engineered direct release path to the blowout panel, and more than two doors must fail for the steam to impact the flood areas containing equipment for keeping offsite power.
Case 4: Impact of high energy line break at MS tunnel	High energy steam is assumed to propagate from the MS tunnel through the turbine hall to the environment. It was assumed in the base case that the components in the turbine hall were failed. In the sensitivity analysis, the components in the turbine hall are not failed since the components are located at much lower elevation than the expected steam propagation pathway.	*Negligible change of CDF and LRF*	The uncertainty associated with this assumption is insignificant.

Table 25.10.3-2 Sensitivity Analyses for Internal Flooding at Power PSA to Quantitatively Assess Modelling Uncertainties (2/2)

Sensitivity Analysis Case	Description	Quantitative Impact	Discussion
Case 5: Selection of BOC pipes in MS Tunnel	The BOC pipes were selected based on the UK ABWR P&IDs for GDA. Since the P&IDs for GDA do not include detailed information, some Non-BOC pipes may be re-allocated to BOC pipes when detailed P&IDs for UK ABWR are available. In order to understand the potential risk impact, detailed P&IDs for an existing ABWR are checked to identify additional BOC pipes.	1 % increase in CDF and LRF	It is concluded that the uncertainty of BOC pipe allocation due to lack of detailed P&IDs during GDA is insignificant.
Case 6: Treatment of S/P drainage in the scenarios in R/B B2F	The elevation of R/B B2F floor is lower than S/P normal water level by 50cm. In the base case, it was conservatively assumed that break of a pipe connected to S/P completely drains S/P by gravity and thus S/P is not credited. A sensitivity analysis is performed to understand the degree of conservatism. If a scenario starting at R/B B2F does not credit S/P in the base case, S/P is credited in the sensitivity analysis case.	Negligible change of CDF and LRF	The reason for the negligible impact is because the flooding and/or high energy steam impacts anyway fail RCIC, HPCF and LPFL in the significant scenarios.
Case 7: Update HEPs based on latest HRA report	Change the probabilities of significant HEPs in the existing CDF and LRF cutsets if base Internal Events HEPs have been updated following the base case quantification of the Flooding PSA. Note that this is a simplified cutset base sensitivity analysis to understand the level of impact.	10 % increase in total CDF 6 % increase in total LRF	The significant increases in CDF and LRF are mostly due to the significant increases of common cognition error HEP for long term heat removal and short term core cooling HEPs. Refinement of these HEPs should be considered.

Table 25.10.3-3 Sensitivity Analyses for Internal Flooding at Power PSA to Quantitatively Assess Design Alternatives

Sensitivity Analysis Case	Description	Quantitative Impact	Discussion
Case 8: Credit flood drains	Floor drain was not credited in the base case due to the limited design/operational information. In the sensitivity analysis, the floor drains are credited such that impact from a spray scenario is limited within the source area.	19 % reduction in CDF 14 % reduction in LRF	Adequate credit to the floor drains is important for realistic risk evaluation, once the design/operational information is available.
Case 9: Sealed penetrations on ceiling of R/B basement floor	Ceiling penetrations in R/B basement floor were assumed to be unsealed. In the sensitivity analysis, these penetrations are assumed to be sealed.	5 % reduction in CDF	The reduction of CDF is not significant.
Case 10: Raised height of Class 1 RPV level transmitters in instrument rack rooms	In the sensitivity analysis, the critical height of Class 1 RPV level transmitters was assumed to be higher than the maximum flood height of the risk significant flood submergence scenarios.	99 % reduction in CDF of a scenario (contributing 5 % to total CDF in base case).	It is very likely that many of the other scenarios would also benefit significantly from preventing these transmitters from failure.

Table 25.10.3-4 Simplified Sensitivity Analyses for SFP Internal Flooding PSA Scoping Analysis to Quantitatively Assess Conservatisms

Sensitivity Analysis Case	Description	Quantitative Impact	Discussion
Consider realistic consequence from PCV failure without core damage (Level 1 PSA success sequences)	PCV failure without core damage (Level 1 PSA success paths with containment failure) is assumed to fail all the systems in R/B with a 1.0 conditional probability due to the potential steaming inside the R/B. In the sensitivity analysis, the conditional probability was reduced to 0.1 by considering that the drywell head flange is likely to fail first given PCV overpressure.	55 % reduction in FDF	This sensitivity analysis shows significant impact by removing the conservatism.
Remove FDF which is double-counting the LRF of reactor	Approximately 32 % of the FDF for SFP POS F involves loss of SFP cooling or SFP leakage induced by containment failure after core damage (large release sequences in Flooding at Power PSA). This fraction of FDF is double-counting the Flooding at Power LRF.	32 % reduction in FDF	This sensitivity analysis shows significant impact by removing the conservatism.

25.10.3.5 Insights from Assessment

From the results, significant internal flood risk insights can be developed. This section discusses these insights.

(1) At Power

- The scenarios originating from R/B contribute 90 percent to the total LRF.
- HELBs in the Main Steam Tunnel have the largest contribution to risk. These scenarios contribute approximately 18 percent to the total CDF and approximately 16 percent to the total LRF, which is a far higher contribution than internal flood scenarios in any other flood area.
- Scenarios that are also relatively important involve those that result in ATWS due to flood impacts on the Reactor Protection System (RPS) signal.

Note: The above insights are superseded by those in (4) at Power (Refinement).

(2) Shutdown (scoping analysis)

Based on this scoping analysis, it was concluded that FDFs during POS B-2 and POS C were sufficiently lower than the at Power LRF. Since POSs B-2 and C are deemed dominant contributors to the FDF due to internal flooding during Shutdown, the internal flood risk during Shutdown is expected to be also insignificant compared to the Internal Flood at Power risk. The specific insights are listed below.

- Flood scenarios inside the PCV are the dominant contributors. This is because the flood water further propagates out of the PCV to the R/B through the open hatches during shutdown.
- Maintenance-induced flood scenarios have insignificant contributions. This is because the maintenance-induced flood scenarios are initiated at the divisions in maintenance where credited components are not impacted unless flood propagates to the divisions in service.

(3) SFP (scoping analysis)

Based on this scoping analysis, the internal flood FDF of SFP at Power is approximately 17 percent of the internal flood LRF for the reactor at Power. This is due to a number of conservatisms in place in the current scoping analysis. By considering the sensitivity analyses in Table 25.10.3-4, the FDF would be much smaller. Furthermore, the component recovery and smaller HEPs would be credited given long time available (more than 300 hours). It is concluded that the FDF of SFP during POS F is also insignificant compared to the at Power flooding risk of reactor.

(4) At Power (Refinement)

- Internal flood risk originating in the R/B contributes approximately 80 percent to the Internal Flood at Power CDF and LRF. This insight is the same as that for the base case.
- Contribution of the flood submergence scenarios to LRF has become higher than that from the HELB scenarios (unlike the base case), since the HELB scenarios were the most extensively refined.

- Contributions of ATWS sequences to CDF and LRF have significantly reduced since manual SCRAM was newly credited.

- Contributions of containment bypass sequences (unisolated BOC/ISLOCA) have significantly reduced since the BOC scenarios were refined.

- Contribution of LOCA inside containment to CDF and LRF is high although only consequential LOCA is considered in IFPSA.

25.10.3.6 Key Assumptions and Study Limitations

Assumptions in the IFPSA were made in the development phase. They relate to each of the tasks defined in Section 25.10.3.2. In addition, when the level of the plant requirements specification limits the detail of the PSA, then appropriate assumptions were developed to allow for completion of the PSA.

Starting from assumptions, sensitivity analyses were performed. Key assumptions, which have a comparatively large impact on the results, were identified from the results of sensitivity analyses. These key assumptions relate to Task 2, Task 7, and the Shutdown and SFP Scoping Analysis, as listed below.

Task 2: Identify Flood Sources, Flood Mechanisms, and SSCs

- Penetrations sealed for fire are assumed not water-tight or steam-tight. This assumption potentially causes significant conservatism.

Task 7: Flood Consequence Analysis

- All the passive components are assumed to fail given high energy line break in the same flood area in the base case. This approach is conservative, and significantly impacts the Flood PSA results as demonstrated by Sensitivity Analysis Case 2 in Table 25.10.3-2. In the refined IFPSA, pipe whip/jet impingement impacts on check valves used by Flooding System of Specific Safety Facility (FLSS) are removed for HELB scenarios in the Main Steam Tunnel Room (MSTR).

- Floor drains are not credited. This approach is conservative, and significantly impacts the Flood PSA results as demonstrated by Sensitivity Analysis Case 8 in Table 25.10.3-3.

- High energy line break inside the secondary containment is assumed to fail all the components inside the secondary containment which are susceptible to damage by high energy steam in the base case. This assumption potentially causes significant conservatism as there should be engineered release pathway of high energy steam (information not available in GDA phase). In the refined IFPSA, flood impact on SSCs in other divisions and solenoid valves for Alternative Rod Insertion (ARI) is removed for Break Outside Containment (BOC) scenarios.

Task 8: Human Reliability Analysis

- In the refined IFPSA, credit of manual isolation within 1 hour was removed. The main reason is complex for diagnosis. Treatment of instrumentation and procedures supporting detection and diagnosis of flooding are one of the uncertainties (conservatism) in GDA phase.

Shutdown and SFP Scoping Analysis

- Local operations inside a building credited in the internal events Shutdown and SFP PSAs, e.g., alignment of manual valves for injection by MUWC or Fire Protection system, are also credited in the scoping analysis.

25.10.4 Level 3 PSA for Internal Hazards

25.10.4.1 Release Category definitions, frequencies and representative conditional consequences

In the Level 2 PSAs for Internal Fire at Power and Internal Flood at Power, the frequency of core melt from each initiator is split between the release categories from the Internal Event At Power (IEAP) Level 2 PSA [Ref-25.10]. This is because the accident severity for the IEAP adequately represents the Internal Fire at Power or Internal Flood at Power initiators.

In the Level 3 PSA, the Internal Fire at Power and Internal Flood at Power release category frequencies are allocated to the equivalent release category from the IEAP Level 2 PSA [Ref-25.10]. Overall, the frequency of fuel melt resulting from Internal Hazards (Fire and Flood) at Power is a factor 10 higher than that from all IEAP. This is due to the conservative nature of the Level 1 / Level 2 PSA for Internal Fire at Power and Internal Flood at Power in the GDA.

25.10.4.2 Internal Fire

The twenty three release categories and their frequencies, brought forward from the Internal Fire at Power Level 2 PSA are summarised in Table 25.10.4-1. Also given is the Level 3 PSA case identifier used to represent the conditional consequences.

It can be seen that the summated frequency for Internal Fire at Power leading to core melt is about a factor of 2 higher than that for all IEAP.

Table 25.10.4-1 Internal Hazards (Fire) Release Categories, Frequencies and Representative Conditional Consequences

Release Categories	Frequency (/y)	Representative Level 3 PSA case
IF 1 Containment Leakage from D/W - failed RPV (TQUV)	2.33E-08	P1
IF 2 Containment Venting (TQUV no DW sprays)	4.99E-10	P2
IF 3 Filtered Containment Venting (TQUV)	1.67E-07	P3
IF 4 Early Containment Failure (AC)	5.47E-08	P4
IF 5-1 Late Containment Failure (TQUV)	9.79E-10	P5-1
IF 5-2 Late Containment Failure (AE)	1.56E-10	P5-2
IF 5-3 Late Containment Failure (AW-LP)	9.58E-10	P5-3
IF 5-4 Late Containment Failure (TW-LP)	9.01E-09	P5-4
IF 6 Late Containment Failure with PCV spray (AE)	1.61E-09	P6
IF 7-1 In-vessel Fuel-Coolant Interaction (TQUV)	1.00E-13	P7-1
IF 7-2 In-vessel Fuel-Coolant Interaction (AE)	1.00E-13	P7-2
IF 8-1 Ex-vessel Fuel-Coolant Interaction (TQUV)	3.10E-11	P8-1
IF 8-2 Ex-vessel Fuel-Coolant Interaction (AE)	3.02E-11	P8-2
IF 9 Direct Containment Heating (TQUX)	2.16E-10	P9
IF 10-1 PCV Isolation failure (TQUV)	8.06E-10	P10-1
IF 10-2 PCV Isolation failure (AE)	7.07E-09	P10-2
IF 11-1 Molten Core Concrete Interaction (TQUV)	1.87E-10	P11-1
IF 11-2 Molten Core Concrete Interaction (AE)	2.01E-11	P11-2
IF 12 RPV rupture (S4)	2.62E-09	P12
IF 13 Containment Bypass (S3E)	7.54E-08	P13
IF 14 S/P Bypass (TNQUV)	1.24E-07	P14
IF 15 Direct Debris Interaction (TQUX)	2.63E-08	P15
IF 16 Long Term SBO (TB in-vessel FCI)	1.30E-10	P16
Total Frequency (/y):	**4.95E-07**	

25.10.4.3 Internal Flood

The twenty three release categories and their frequencies, brought forward from the Internal Flood at Power Level 2 PSA are summarised in Table 25.10.4-2. Also given is the Level 3 PSA case used to represent the conditional consequences.

It can be seen that the summated frequency for Internal Fire at Power leading to core melt is about a factor of 7 higher than that for all IEAP.

Table 25.10-4-2 Internal Hazards (Flood) Release Categories, Frequencies and Representative Conditional Consequences

Release Categories	Frequency (/y)	Representative Level 3 PSA case
IFL 1 Containment Leakage from D/W - failed RPV (TQUV)	7.71E-09	P1
IFL 2 Containment Venting (TQUV no DW sprays)	7.84E-09	P2
IFL 3 Filtered Containment Venting (TQUV)	1.16E-06	P3
IFL 4 Early Containment Failure (AC)	3.73E-08	P4
IFL 5-1 Late Containment Failure (TQUV)	2.58E-08	P5-1
IFL 5-2 Late Containment Failure (AE)	2.40E-09	P5-2
IFL 5-3 Late Containment Failure (AW-LP)	3.28E-09	P5-3
IFL5-4 late Containment Failure (TW-LP)	2.65E-08	P5-4
IFL 6 Late Containment Failure with PCV spray (AE)	4.33E-08	P6
IFL7-1 In-vessel Fuel-Coolant Interaction (TQUV)	2.38E-11	P7-1
IFL7-2 In-vessel Fuel-Coolant Interaction (AE)	4.04E-12	P7-2
IFL8-1 Ex-vessel Fuel-Coolant Interaction (TQUV)	3.26E-09	P8-1
IFL8-2 Ex-vessel Fuel-Coolant Interaction (AE)	5.49E-10	P8-2
IFL 9 Direct Containment Heating (TQUX)	2.40E-09	P9
IFL 10-1 PCV Isolation failure (TQUV)	5.13E-08	P10-1
IFL 10-2 PCV Isolation failure (AE)	5.47E-08	P10-2
IFL11-1 Molten Core Concrete Interaction (TQUV)	8.74E-09	P11-1
IFL11-2 Molten Core Concrete Interaction (AE)	5.05E-10	P11-2
IFL 12 RPV rupture (S4)	4.39E-08	P12
IFL 13 Containment Bypass (S3E)	2.74E-08	P13
IFL 14 S/P Bypass (TNQUV)	2.44E-09	P14
IFL 15 Direct Debris Interaction (TQUX)	2.36E-07	P15
IFL 16 Long Term SBO (TB in-vessel FCI)	9.24E-12	P16
Total Frequency (/y):	**1.75E-06**	

25.10.4.4 Conditional Consequences for Level 3 PSA

In the Level 2 PSA, the frequency of core melt is split between the release categories for IEAP [Ref-25.10]. In the Level 3 PSA reported here, the conditional consequences for the equivalent IEAP release categories from [Ref-25.1] are used to represent the conditional consequences. The Level 3 PSA case identifiers are given in Tables 25.10.4-1 and Table 25.10.4-2.

The conditional consequences for these Level 3 PSA cases are given in Section 25.6.

It is considered that there may be some conservatism in the derivation of frequency for these release categories in the GDA. Therefore, these results should be considered as a conservative assessment of the contribution of internal hazards to the combined risk profile. It is expected that this will be reduced in later revisions of the Internal Hazards PSAs in the post GDA phase.

25.10.4.5 Facility Dose Bands

The summated assessment for Internal Hazards: Fire and Flood at Power leading to fuel melt is given in Table 25.10.4-3. The summated frequency for each facility dose band is presented to enable comparison with the BSO and BSL for Target 8.

- There is no contribution to the 0.0001 to 0.001 Sv (0.1 to 1 mSv) and 0.01 to 0.1 Sv (10 to 100 mSv) facility dose bands.
- There is a negligible contribution to the 0.001 to 0.01 Sv (1 to 10 mSv) facility dose band, at <0.01 percent of the BSO.
- Release category 3 (identifiers IF 3 and IFL 3) contributes 1.33E-06 /y to the 0.1 to 1 Sv (100 to 1,000 mSv) facility dose band. This is equivalent to 13.3 percent of the BSO, with Internal fire contributing 1.67E-07 /y and Internal Flood 1.16E-06 /y.
- The remaining release categories contribute 8.82E-07 /y to the >1 Sv (>1,000 mSv) facility dose band. This is equivalent to 88.2 percent of the BSO.

Table 25.10.4-3 Assessment against Facility Dose Bands (Target 8) for Internal Hazards: Fire and Flood at Power Leading to Fuel Melt

Facility Dose Band (Sv)	Release categories assigned to each dose band for IEAP leading to fuel melt	Band Frequency (/y)	Percentage of Band BSO	BSO (/y)	BSL (/y)
0.0001 to 0.001	-	-	-	1.0E-2	1
0.001 to 0.01	IF1, IFL1	3.10E-08	0.0 %	1.0E-3	1.0E-1
0.01 to 0.1	-	-	-	1.0E-4	1.0E-2
0.1 to 1	IF3, IFL3	1.33E-06	13.3 %	1.0E-5	1.0E-3
> 1	IF2, IFL2, IF4, IF5-1, IF5-2, IF5-3, IF5-4, IF6, IF7-1, IF7-2, IF8-1, IF8-2, IF9, IF10-1, IF10-2, IF11-1, IF11-2, IF12, IF13, IF14, IF15, IF16, IFL4, IFL5-1, IFL5-2, IFL5-3, IFL5-4, IFL6, IFL7-1, IFL7-2, IFL8-1, IFL8-2, IFL9, IFL10-1, IFL10-2, IFL11-1, IFL11-2, IFL12, IFL13, IFL14, IFL15, IFL16	8.82E-07	88.2 %	1.0E-6	1.0E-4
Summated frequency of release categories /y		2.24E-06			

Contribution from Internal Fire leading to fuel melt

Table 25.10.4-4 presents the assessment against Target 8 for the Internal Hazards: Fire at Power, based on the release category frequencies of Table 25.10.4-1 and the facility dose band allocations.

The Internal Fire initiators alone contribute 3.05E-07 /y to the >1 Sv (>1,000 mSv) facility dose band. This is equivalent to 30.5 percent of the BSO.

- Four release categories contribute 2.80E-07 /y or 92.0 percent of the total from Internal Fire and, individually, are > 1 percent of the BSO for this dose band:
  - Release category IF 14, S/P Bypass, at 1.24E-07 /y (40.7 percent of the band frequency).
  - Release category IF 13 Containment Bypass (S3E), at 7.54E-08 /y (24.7 percent of the band frequency).
  - Release category IF 4 Early Containment Failure (AC), at 5.47E-08 /y (18.0 percent of the band frequency),
  - Release category IF 15 Direct Debris Interaction (TQUX), at 2.63E-08 /y (8.6 percent of the band frequency),

Table 25.10.4-4 Assessment against Facility Dose Bands (Target 8) for Internal Hazards: Fire at Power Leading to Fuel Melts

Facility Dose Band (Sv)	Release categories assigned to each dose band for IEAP leading to fuel melt	Band Frequency (/y)	Percentage of Band BSO	BSO (/y)	BSL (/y)
0.0001 to 0.001	-	-	-	1.0E-2	1
0.001 to 0.01	IF1	2.33E-08	0.0 %	1.0E-3	1.0E-1
0.01 to 0.1	-	-	-	1.0E-4	1.0E-2
0.1 to 1	IF3	1.67E-07	1.7 %	1.0E-5	1.0E-3
> 1	IF2, IF4, IF5-1, IF5-2, IF5-3, IF5-4, IF6, IF7-1, IF7-2, IF8-1, IF8-2, IF9, IF10-1, IF10-2, IF11-1, IF11-2, IF12, IF13, IF14, IF15, IF16	3.05E-07	30.5 %	1.0E-6	1.0E-4
Summated frequency of release categories /y		4.95E-07			

Contribution from Internal Flood leading to fuel melt

Table 25.10.4-5 presents the assessment against Target 8 for the Internal Hazards: Flood at Power, based on the release category frequencies of Table 25.10.4-2 and the facility dose band allocations presented.

The Internal Flood initiators alone contribute 5.78E-07 /y to the >1 Sv (>1,000 mSv) facility dose band. This is equivalent to 57.8 percent of the BSO.

- Release category IFL 15, Direct Debris Interaction (TQUX), is dominant and contributes 2.36E-07 /y (40.8 percent of the band frequency).
- A further eight release categories contribute an additional 3.10E-07 /y or 53.7 percent of the total and, individually, are >1 percent of the BSO for this dose band:
  - Release category IFL10-2 PCV Isolation Failure (AE), at 5.47E-08 /y (9.5 percent of the band frequency),
  - Release category IFL10-1 PCV Isolation Failure (TQUV), at 5.13E-08 /y (8.9 percent of the band frequency).
  - Release category IFL12 RPV rupture (S4), at 4.39E-08 /y (7.6 percent of the band frequency).
  - Release category IFL6 Late Containment Failure with PCV spray (AE), at 4.33E-08 /y (7.5 percent of the band frequency),
  - Release category IFL4 Early Containment Failure (AC), at 3.73E-08 /y (6.5 percent of the band frequency),
  - Release category FL13 Containment Bypass (S3E), at 2.74E-08 /y (4.7 percent of the band total),
  - Release category IFL5-4 late Containment Failure (TW-LP), at 2.65E-08 /y (4.6 percent of the band frequency),
  - Release category IFL5-1 Late Containment Failure (TQUV), at 2.58E-08 /y (4.5 percent of the band frequency),

Table 25.10.4-5 Assessment against Facility Dose Bands (Target 8) for Internal Hazards: Flood at Power Leading to Fuel Melt

Facility Dose Band (Sv)	Release categories assigned to each dose band for IEAP leading to fuel melt	Band Frequency (/y)	Percentage of Band BSO	BSO (/y)	BSL (/y)
0.0001 to 0.001	-	-	-	1.0E-2	1
0.001 to 0.01	IFL1	7.71E-09	0.0 %	1.0E-3	1.0E-1
0.01 to 0.1	-	-	-	1.0E-4	1.0E-2
0.1 to 1	IFL3	1.16E-06	11.6 %	1.0E-5	1.0E-3
> 1	IFL2, IFL4, IFL5-1, IFL5-2, IFL5-3, IFL5-4, IFL6, IFL7-1, IFL7-2, IFL8-1, IFL8-2, IFL9, IFL10-1, IFL10-2, IFL11-1, IFL11-2, IFL12, IFL13, IFL14, IFL15, IFL16	5.78E-07	57.8 %	1.0E-6	1.0E-4
Summated frequency of release categories /y		1.75E-06			

25.10.4.6 Individual Risk

The individual risk for each release category for Internal Hazards: Fire and Flood at Power leading to fuel melt is calculated as the product of the conditional individual risk for the equivalent IEAP release category given in Section 25.6 and the release category frequency given in Table 25.10.4-1 or Table 25.10.4-2.

The summated individual risk at 1 km from Internal Hazards at Power is 6.38E-08 /y or 6.4 percent of the BSO. This is shown in Table 25.10.4-6.

Table 25.10.4-6 Individual Risk of Fatality Close to the Site (Target 7) for Internal Hazards Fire and Flood at Power Leading to Fuel Melt

Fault Group	Individual risk of fatal health effects (/y)		
	400 m	**1,000 m**	**1,500 m**
Internal Hazards: Fire at Power	4.73E-08	2.99E-08	2.34E-08
Internal Hazards: Flood at Power	6.65E-08	3.39E-08	2.46E-08
Total individual risk (/y):	1.14E-07	6.38E-08	4.80E-08
Total as % of BSO	**11.4 %**	**6.4 %**	**4.8 %**
BSO	**1.00E-06**		
BSL	**1.00E-04**		

Contribution from Internal Fire leading to fuel melt

The individual risk for each release category for Internal Hazards: Fire at Power leading to fuel melt is calculated as the product of the conditional individual risk for the equivalent IEAP release category and the release category frequency given in Table 25.10.4-1. The contribution of each release category to the overall risk from Internal Hazards: Fire at Power leading to fuel melt is given in Table 25.10.4-7 and represented graphically in Figure 25.10.4-1.

The Internal Fire initiators alone contribute 2.99E-08 /y to the individual risk at 1km. This is equivalent to 3.0 percent of the BSO.

- Five release categories contribute 2.71E-08 /y or 90.45 percent of the total from Internal Fire and, individually, are >1 percent of the BSO at 1km:
  - Release category IF14 S/P Bypass (TNQUV), at 1.17E-08 /y (39.01 percent of the total).
  - Release category IF13 Containment Bypass (S3E), at 9.28E-09 /y (31.07 percent of the total).
  - Release category IF4 Early Containment Failure (AC), at 6.09E-09 /y (20.37 percent of the total).

Table 25.10.4-7 Individual Risk of Fatality Close to the Site (Target 7) for Internal Hazards: Fire at Power Leading to Fuel Melt

Release Category	Release Category Frequency (/y)	Individual risk of fatal health effects (/y)			Contribution to Total Risk at 1 km
		400 m	1,000 m	1,500 m	
IF1	2.33E-08	1.36E-13	5.33E-14	3.30E-14	0.00 %
IF2	4.99E-10	7.04E-12	2.04E-12	1.13E-12	0.01 %
IF3	1.67E-07	1.24E-10	4.39E-11	2.86E-11	0.15 %
IF4	5.47E-08	9.58E-09	6.09E-09	4.87E-09	20.37 %
IF5-1	9.79E-10	7.19E-11	2.97E-11	1.96E-11	0.10 %
IF5-2	1.56E-10	2.33E-11	1.22E-11	8.34E-12	0.04 %
IF5-3	9.58E-10	2.11E-10	1.39E-10	1.10E-10	0.46 %
IF5-4	9.01E-09	9.73E-10	3.97E-10	2.56E-10	1.33 %
IF6	1.61E-09	1.25E-10	4.50E-11	2.88E-11	0.15 %
IF7-1	1.00E-13	1.69E-14	9.31E-15	6.61E-15	0.00 %
IF7-2	1.00E-13	1.91E-14	1.21E-14	9.29E-15	0.00 %
IF8-1	3.10E-11	1.51E-12	4.89E-13	2.97E-13	0.00 %
IF8-2	3.02E-11	4.65E-12	2.54E-12	1.77E-12	0.01 %
IF9	2.16E-10	2.71E-11	1.18E-11	8.03E-12	0.04 %
IF10-1	8.06E-10	5.31E-11	1.86E-11	1.16E-11	0.06 %
IF10-2	7.07E-09	1.22E-09	7.98E-10	6.24E-10	2.67 %
IF11-1	1.87E-10	1.52E-11	5.58E-12	3.47E-12	0.02 %
IF11-2	2.01E-11	3.54E-12	1.81E-12	1.25E-12	0.01 %
IF12	2.62E-09	4.62E-10	2.77E-10	2.03E-10	0.93 %
IF13	7.54E-08	1.26E-08	9.28E-09	7.99E-09	31.07 %
IF14	1.24E-07	1.92E-08	1.17E-08	8.58E-09	39.01 %
IF15	2.63E-08	2.56E-09	1.06E-09	6.82E-10	3.53 %
IF16	1.30E-10	2.19E-11	1.41E-11	1.09E-11	0.05 %
Total individual risk (/y):		**4.73E-08**	**2.99E-08**	**2.34E-08**	
Total as % of BSO		**4.7 %**	**3.0 %**	**2.3 %**	
BSO		**1.00E-06**			
BSL		**1.00E-04**			

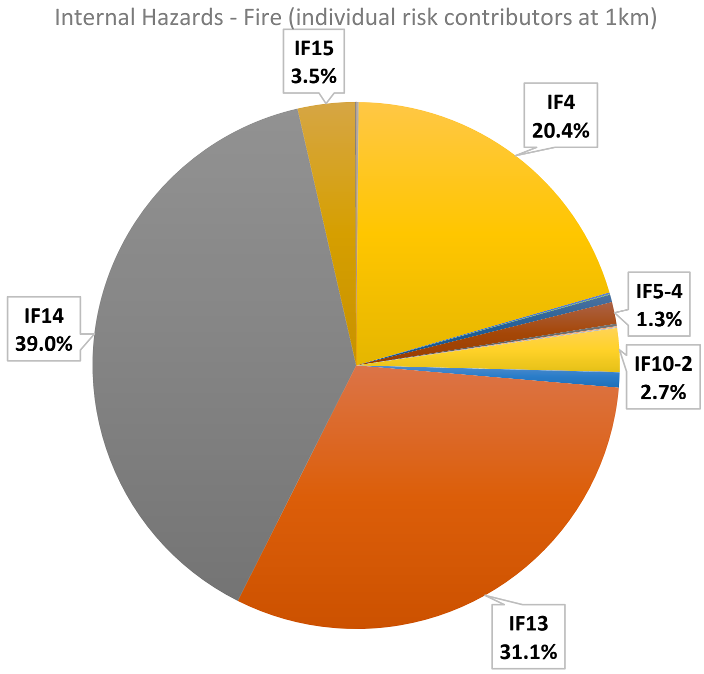


Figure 25.10.4-1 Contribution of Release Categories to the Individual Risk at 1 km for Internal Hazards: Fire Leading to Fuel Melts

Contribution from Internal Flood leading to fuel melt

The individual risk for each release category for Internal Hazards: Flood at Power leading to fuel melt is calculated as the product of the conditional individual risk for the equivalent IEAP release category and the release category frequency given in Table 25.10.4-2. The contribution of each release category to the overall risk from Internal Hazards: Flood at Power leading to fuel melt is given in Table 25.10.4-8 and represented graphically in Figure 25.10.4-2.

The Internal Flood initiators alone contribute 3.39E-08 /y to the individual risk at 1km. This is equivalent to 3.4 percent of the BSO.

- Release category IFL 15, Direct Debris Interaction (TQUX), is dominant and contributes 9.46E-09 /y (27.9 percent of total). This is due to the combination of a significant frequency and high conditional consequences.
- All other release categories individually contributes to less than 1 percent of the BSO at 1km.

Table 25.10.4-8 Individual Risk of Fatality Close to the Site (Target 7) for Internal Hazards: Flood at Power Leading to Fuel Melt

Release Category	Release Category Frequency (/y)	Individual risk of fatal health effects (/y)			Contribution to Total Risk at 1 km
		400 m	1,000 m	1,500 m	
IFL1	7.71E-09	4.50E-14	1.76E-14	1.09E-14	0.00 %
IFL2	7.84E-09	1.11E-10	3.20E-11	1.78E-11	0.09 %
IFL3	1.16E-06	8.59E-10	3.04E-10	1.99E-10	0.90 %
IFL4	3.73E-08	6.53E-09	4.15E-09	3.32E-09	12.25 %
IFL5-1	2.58E-08	1.90E-09	7.84E-10	5.16E-10	2.31 %
IFL5-2	2.40E-09	3.59E-10	1.87E-10	1.28E-10	0.55 %
IFL5-3	3.28E-09	7.22E-10	4.76E-10	3.78E-10	1.40 %
IFL5-4	2.65E-08	2.86E-09	1.17E-09	7.53E-10	3.45 %
IFL6	4.33E-08	3.35E-09	1.21E-09	7.75E-10	3.57 %
IFL7-1	2.38E-11	4.03E-12	2.22E-12	1.57E-12	0.01 %
IFL7-2	4.04E-12	7.72E-13	4.91E-13	3.76E-13	0.00 %
IFL8-1	3.26E-09	1.58E-10	5.14E-11	3.13E-11	0.15 %
IFL8-2	5.49E-10	8.45E-11	4.61E-11	3.23E-11	0.14 %
IFL9	2.40E-09	3.00E-10	1.31E-10	8.90E-11	0.39 %
IFL10-1	5.13E-08	3.38E-09	1.18E-09	7.38E-10	3.49 %
IFL10-2	5.47E-08	9.47E-09	6.17E-09	4.83E-09	18.21 %
IFL11-1	8.74E-09	7.11E-10	2.61E-10	1.63E-10	0.77 %
IFL11-2	5.05E-10	8.88E-11	4.54E-11	3.12E-11	0.13 %
IFL12	4.39E-08	7.73E-09	4.63E-09	3.40E-09	13.65 %
IFL13	2.74E-08	4.58E-09	3.38E-09	2.91E-09	9.96 %
IFL14	2.44E-09	3.79E-10	2.30E-10	1.69E-10	0.68 %
IFL15	2.36E-07	2.30E-08	9.46E-09	6.11E-09	27.90 %
IFL16	9.24E-12	1.56E-12	1.00E-12	7.74E-13	0.00 %
Total individual risk (/y):		**6.65E-08**	**3.39E-08**	**2.46E-08**	
Total as % of BSO		**6.7 %**	**3.4 %**	**2.5 %**	
BSO		**1.00E-06**			
BSL		**1.00E-04**			

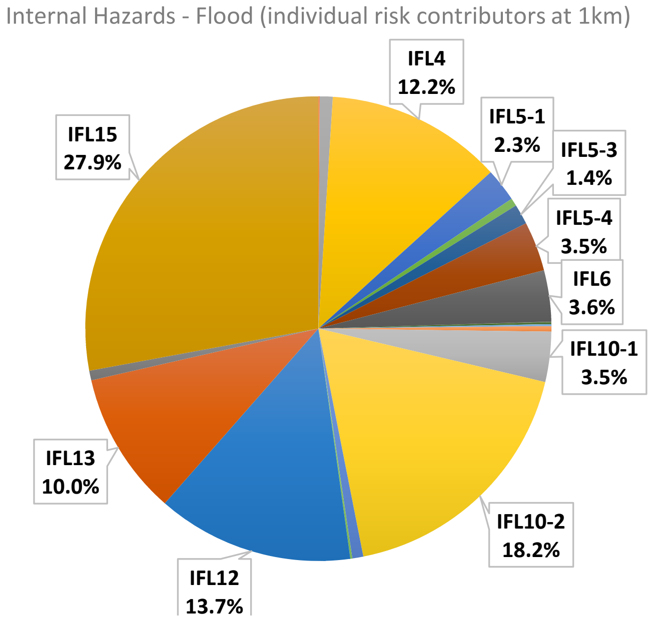


Figure 25.10.4-2 Contribution of Release Categories to the Individual Risk at 1 km for Internal Hazards: Flood Leading to Fuel Melts

25.10.4.7 Societal Risk

Table 25.10.4-9 presents the assessment against Target 9 for the Internal Hazards: Fire and Flood at Power leading to fuel melt. This is based on the summated numbers of short term fatal health effects and notional late fatalities in the UK population presented in Section 25.6.

All release categories except IF 1, IF 3, IFL 1 and IFL 3 are assigned as above the Target 9 threshold.

The summated societal risk from Internal Hazards: Fire and Flood at Power is 8.82E-07 /y, i.e. about a factor of 9 above the BSO. This is equivalent to 9 percent of the BSL. This is considered to be in the tolerable if ALARP region; however, it is noted that there is conservatism in the derivation of Internal Hazard release category frequencies for GDA.

It can be seen from Table 25.10.4-10 and Table 25.10.4-11 that two release categories are individually above the BSO.

Table 25.10.4-9 Frequency of Exceeding the Societal Threshold (Target 9) for Internal Hazards Fire and Flood at Power Leading to Fuel Melt

Sum of all release categories	Frequency above Target 9 threshold (/y)
Internal Hazards: Fire at Power	**3.05E-07**
Internal Hazards: Flood at Power	**5.78E-07**
Total frequency (/y):	**8.82E-07**
BSO	**1.00E-07**
Total as % of BSL:	**8.8 %**
BSL	**1.00E-05**

Table 25.10.4-10 Frequency of Exceeding the Societal Threshold (Target 9) for Internal Hazards: Fire leading to Fuel Melts with Minimal Offsite Protective Actions

Release Category	Release category frequency (/y)	Frequency above Target 9 threshold (/y)	Contribution to Total Frequency
IF1	2.33E-08	0.00E+00	0.00 %
IF2	4.99E-10	4.99E-10	0.16 %
IF3	1.67E-07	0.00E+00	0.00 %
IF4	5.47E-08	5.47E-08	17.96 %
IF5-1	9.79E-10	9.79E-10	0.32 %
IF5-2	1.56E-10	1.56E-10	0.05 %
IF5-3	9.58E-10	9.58E-10	0.31 %
IF5-4	9.01E-09	9.01E-09	2.96 %
IF6	1.61E-09	1.61E-09	0.53 %
IF7-1	1.00E-13	1.00E-13	0.00 %
IF7-2	1.00E-13	1.00E-13	0.00 %
IF8-1	3.10E-11	3.10E-11	0.01 %
IF8-2	3.02E-11	3.02E-11	0.01 %
IF9	2.16E-10	2.16E-10	0.07 %
IF10-1	8.06E-10	8.06E-10	0.26 %
IF10-2	7.07E-09	7.07E-09	2.32 %
IF11-1	1.87E-10	1.87E-10	0.06 %
IF11-2	2.01E-11	2.01E-11	0.01 %
IF12	2.62E-09	2.62E-09	0.86 %
IF13	7.54E-08	7.54E-08	24.74 %
IF14	1.24E-07	1.24E-07	40.67 %
IF15	2.63E-08	2.63E-08	8.65 %
IF16	1.30E-10	1.30E-10	0.04 %
	Total Frequency: /y	**3.05E-07**	
	BSO	**1.00E-07**	
	Total as % of BSL	**3.1 %**	
	BSL	**1.00E-05**	

Table 25.10.4-11 Frequency of Exceeding the Societal Threshold (Target 9) for Internal Hazards: Flood leading to Fuel Melts with Minimal Offsite Protective Actions

Release Category	Release category frequency (/y)	Frequency above Target 9 threshold (/y)	Contribution to Total Frequency
IFL1	7.71E-09	0.00E+00	0.00 %
IFL2	7.84E-09	7.84E-09	1.36 %
IFL3	1.16E-06	0.00E+00	0.00 %
IFL4	3.73E-08	3.73E-08	6.46 %
IFL5-1	2.58E-08	2.58E-08	4.47 %
IFL5-2	2.40E-09	2.40E-09	0.41 %
IFL5-3	3.28E-09	3.28E-09	0.57 %
IFL5-4	2.65E-08	2.65E-08	4.59 %
IFL6	4.33E-08	4.33E-08	7.49 %
IFL7-1	2.38E-11	2.38E-11	0.00 %
IFL7-2	4.04E-12	4.04E-12	0.00 %
IFL8-1	3.26E-09	3.26E-09	0.57 %
IFL8-2	5.49E-10	5.49E-10	0.09 %
IFL9	2.40E-09	2.40E-09	0.42 %
IFL10-1	5.13E-08	5.13E-08	8.88 %
IFL10-2	5.47E-08	5.47E-08	9.47 %
IFL11-1	8.74E-09	8.74E-09	1.51 %
IFL11-2	5.05E-10	5.05E-10	0.09 %
IFL12	4.39E-08	4.39E-08	7.60 %
IFL13	2.74E-08	2.74E-08	4.75 %
IFL14	2.44E-09	2.44E-09	0.42 %
IFL15	2.36E-07	2.36E-07	40.85 %
IFL16	9.24E-12	9.24E-12	0.00 %
Total Frequency: /y		**5.78E-07**	
BSO		**1.00E-07**	
Total as % of BSL		**5.8 %**	
BSL		**1.00E-05**	

Contribution from Internal Fire leading to fuel melt

Table 25.10.4-10 presents the assessment against Target 9 for the Internal Hazards: Fire at Power, based on the release category frequencies of Table 25.10.4-1 and the predicted societal consequences in Section 25.6. The release categories contributing to this frequency are shown in Figure 25.10.4-3.

The Internal Fire initiators alone contribute 3.05E-07 /y to the frequency of exceeding the Target 9 threshold with minimal protective actions. This is equivalent to a factor of 3 above the BSO. This is equivalent to 3.1 percent of the BSL.

Four release categories contribute 2.80E-07 /y or 92.02 percent of the total from Internal Fire and, individually, are above the BSO for Target 9:

- Release category IF14 S/P Bypass (TNQUV), at 1.24E-07 /y (40.67 percent of the total frequency).
- Release category IF13 Containment Bypass (S3E), at 7.54E-08 /y (24.74 percent of the total frequency).
- Release category IF4 Early Containment Failure (AC), at 5.47E-08 /y (17.96 percent of the total frequency),
- Release category IF15 Direct Debris Interaction (TQUX), at 2.63E-08 /y (8.65 percent of the total frequency),

Contribution from Internal Flood leading to fuel melts

Table 25.10.4-11 presents the assessment against Target 9 for the Internal Hazards: Flood at Power, based on the release category frequencies of Table 25.10.4-2 and the predicted societal consequences in Section 25.6. The release categories contributing to this frequency are shown in Figure 25.10.4-4.

The Internal Flood initiators alone contribute 5.78E-07 /y to the frequency of exceeding the Target 9 threshold with minimal protective actions. This is equivalent to a factor of about 9 above the BSO. This is equivalent to 5.8 percent of the BSL.

Release category IFL 15, Direct Debris Interaction, is dominant and contributes 2.36E-07 /y (40.8 percent of the total frequency). This is above the BSO for Target 9.

Other release categories are, individually, less than the BSO for Target 9. The following release categories are, individually above 10 percent of the BSO:

- SIFL10-2, PCV Isolation Failure (AE) at 5.47E-08/y (9.5 percent of the total frequency),
- IFL10-1, PCV Isolation Failure (TQUV) at 5.13E-08/y (8.9 percent of the total frequency),
- IFL12, RPV Rupture (S4) at 3.73E-08/y (7.6 percent of the total),
- IFL6, Late Containment Failure with PCV Sprays (AE) at 4.33E-08/y (7.5 percent of the total frequency),
- IFL4, Early Containment Failure (AC) at 3.73E-08/y (6.5 percent of the total frequency),
- IFL13, Containment Bypass (S3E) at 2.74E-08/y (4.8 percent of the total frequency),
- IFL5-4, Late Containment Failure (TW-LP) at 2.65E-08/y (4.6 percent of the total frequency),

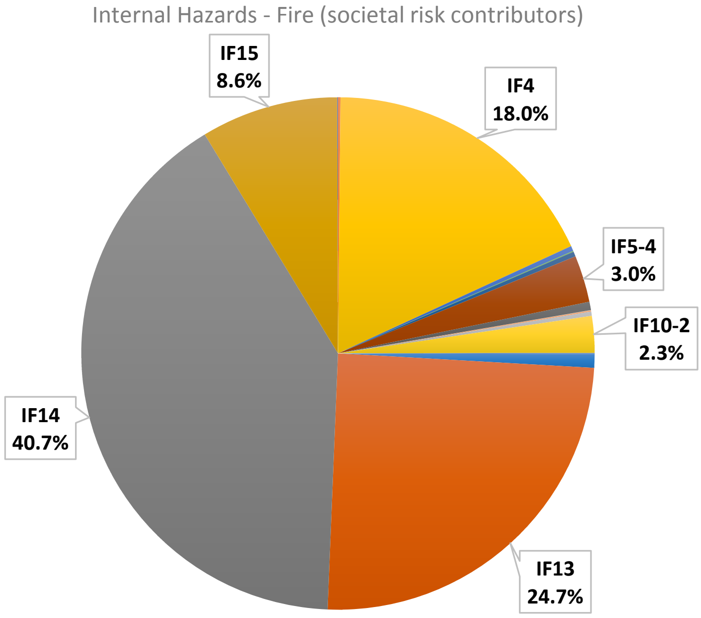


Figure 25.10.4-3 Contribution of Release Categories to the Frequency of Exceeding the Societal Risk Criterion for Internal Hazards: Fire Leading to Fuel Melts

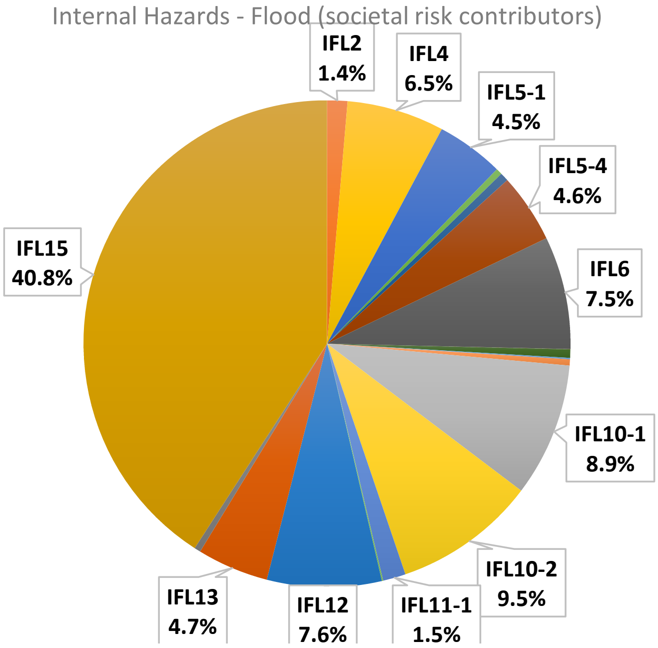


Figure 25.10.4-4 Contribution of Release Categories to the Frequency of Exceeding the Societal Risk Criterion for Internal Hazards: Flood Leading to Fuel Melts

25.10.5 Other Internal Hazard PSA

25.10.5.1 Assessment of Turbine Missiles

This section describes the PSA for the turbine missile to identify potential vulnerabilities in UK ABWR. The PSA in this section is used to demonstrate that the generic design meets Targets 7, 8, and 9 for GDA and to inform the design organizations of potential vulnerabilities to guide future activities, including the support of ALARP demonstration.

In order to identify the external hazards and internal hazards that are risk significant and necessary to consider in the assessment of risk for GDA of the UK ABWR, the screening and prioritisation analyses of Internal Hazard (IH) were presented in Section 25.10.1.

(1) Scope

The Core Damage Frequency (CDF) and Large Release Frequency (LRF) of UK ABWR for turbine missile are assessed. The initiating event frequencies and the postulated impacts are defined. The CCDPs and CLRPs are assessed based on the results of the Internal Events At Power (IEAP) Level 1 PSA [Ref-25.8] and Level 2 PSA [Ref-25.74], respectively.

(2) Methodology

CDF and LRF of each scenario are assessed by multiplying the associated Hazard-Induced Initiating Event (IE) frequency by the CCDP and CLRP, respectively. This section describes the methodologies to derive the CCDPs and CLRP. The postulated impacts from turbine missiles are loss of specific buildings or divisional areas, as well as LOOP. The CCDPs and CLRPs are calculated by the following process:

(a) Prepare cutset files for CDF and LRF which include all the cutsets for the specific IE TEW4 (Weather-related LOOP > 14h), by merging the relevant cutset files generated by the sequence by sequence quantification for Level 1 PSA [Ref-25.8] and Level 2 PSA [Ref-25.74] with a truncation 5.0E-17 /y.

(b) Determine the surrogate basic events to represent the postulated loss of a building or a divisional area.

(c) Set the IE basic event (TEW4) to TRUE in order to derive CCDP and CLRP in the cutset file.

(d) Set the basic events mapped to the specific building/division to TRUE in the cutset file.

(e) Minimise (subsume) the cutset file.

(f) If a Level 1 PSA event tree heading is predetermined to fail due to the treatment in step 4), set the Level 1 PSA sequence flag events to FALSE for the sequences which include success of that heading, in order to avoid unintended non-minimal cutsets.

(3) Assessment

Event scenarios

The impacts on the SSCs from turbine missile hazards are defined [Ref-25.1]. CCDPs and CLRPs are assessed for the cases where a turbine missile impacts one of the following buildings/structures, then CDFs and LRFs are evaluated.

Case I

A turbine missile hits C/B and impacts one of the divisions in C/B. The divisions are I, II, III, IV and Main Control Room (MCR). CCDP and CLRP given loss of each division or the MCR are calculated and the highest CCDP and CLRP are used for the CDF and LRF evaluations, respectively.

Case II

A turbine missile hits R/B and impacts one of the divisions in R/B. The divisions are I, II, III, IV. Although there is secondary containment inside the R/B, both inside and outside the secondary containment are assumed to be impacted. CCDP and CLRP given loss of each division are calculated and the highest CCDP and CLRP are used for the CDF and LRF evaluations, respectively.

Case III

A turbine missile hits B/B and impacts entire B/B (all B/B equipment).

Case IV

A turbine missile hits Rad-Waste Building (Rw/B) and impacts whole Rw/B. Since loss of Rw/B has no impact on the SSCs credited in the internal events PSA, the consequence is just un-recovered LOOP in terms of the reactor PSA. Therefore, the CCDP and CLRP in the tornado missile Case I (un-recovered LOOP only) are used.

Case V

A turbine missile hits Heat Exchanger Building (Hx/B) and impacts one of the divisions in Hx/B. The divisions are I, II and III. CCDP and CLRP given loss of each division are calculated and the highest CCDP and CLRP are used for the CDF and LRF evaluations, respectively.

Case VI

A turbine missile hits one of three EDG Buildings (EDG/Bs) and impacts it. CCDP and CLRP given loss of each EDG/B are calculated. All of them are used for the CDF and CLRP evaluations because the EDG/Bs are the separate buildings and have different frequencies of turbine missile impact.

Case VII

A turbine missile hits the CST.

Evaluation of CCDPs

The CCDPs and CLRPs are calculated by the process introduced in Section 25.10.5.1 (2).

Case I: Turbine missile impacting C/B

The calculated CCDP and CLRP are 4.8E-05and 1.0E-05, respectively. The highest CCDP and CLRP come from loss of Div. I and loss of MCR, respectively. They are used for the CDF and LRF evaluations.

Case II: Turbine missile impacting R/B

The calculated CCDP and CLRP are 6.4E-03 and 4.0E-04, respectively. The highest CCDP and CLRP come from loss of Div. I. They are used for the CDF and LRF evaluations.

Case III: Turbine missile impacting B/B

The CCDP given loss of the B/B is calculated. Given loss of the B/B, the following event tree headings always fail (with the probability of 1.0) due to the dependence on Class 2 AC power and/or B/B DC power.

BBG, D-ADS, FLSS, FLSS-AUTO, DC2, SLC

The calculated CCDP and CLRP are 8.8E-04 and 5.7E-04, respectively.

Case IV: Turbine missile impacting Rw/B

A turbine missile hits Rad-Waste Building (Rw/B) and impacts whole Rw/B. Since loss of Rw/B has no impact on the SSCs credited in the internal events PSA, the consequence is just un-recovered LOOP in terms of the reactor PSA. Therefore, the CCDP and CLRP in the tornado missile Case I (un-recovered LOOP only) are used.

The calculated CCDP and CLRP are 5.3E-06 and 1.2E-06, respectively.

Case V: Turbine missile impacting Hx/B

The calculated CCDP and CLRP are 4.6E-05 and 7.9E-06, respectively. The highest CCDP and CLRP come from loss of Div. II and are used for the CDF and LRF evaluations.

Case VI: Turbine missile impacting EDG/B

The CCDPs given loss of EDG/B-A, EDG/B-B or EDG/B-C are calculated:

Loss of EDG/B-A: CCDP = 3.3E-05 CLRP = 7.0E-06

Loss of EDG/B-B: CCDP = 3.8E-05 CLRP = 7.0E-06

Loss of EDG/B-C: CCDP = 4.0E-05 CLRP = 5.4E-06

Case VII: Turbine missile impacting CST

The calculated CCDP and CLRP are 5.4E-06 and 1.3E-06, respectively.

Evaluation of CDFs and LRFs

Based on the IE frequencies [Ref-25.1] and the associated CCDP and CLRP evaluated above, the CDF and LRF from each target case are calculated. The total CDF and LRF for turbine missile hazards are 7.1E-10 /y and 8.1E-11 /y, respectively.

(4) Uncertainty and Sensitivity Analysis

The total CDF and LRF for turbine missile hazards are 7.1E-10 /y and 8.1E-11 /y, respectively by using the simplified method. Since it is confirmed that these are three orders magnitude lower than the CDF and LRF from the internal initiating events at Power (presented in sections 25.7), the uncertainty and sensitivity analyses was not performed in GDA PSA.

(5) Insights from Assessment

The quantification results provide the following insights:

- The Hx/B is inside the Low Trajectory missile strike zones and thus the IE frequency is much higher than for any other cases.
- CCDP is the highest given the loss of R/B by assuming loss of R/B Div. I in which RCIC, LPFL-A, FLSS, FLSR and all the RHR trains (for containment heat removal) are assumed to fail.
- CLRP is the highest given the loss of B/B because the loss of the B/B impacts ARI, SLC, FLSS, RDCF and manual containment venting system.
- Summation of these CDFs is 7.1E-10 /y. This is three orders of magnitude lower than the CDF from internal initiating events at Power. Contribution form the turbine missile impacting R/B is the highest (69 percent) due to the highest CCDP by assuming loss of R/B Div. I. Hx/B is the second highest (27 percent) contributor because the loss of Hx/B has much higher IE frequency than any other cases.
- Summation of these LRFs is 8.1E-11 /y. This is three orders of magnitude lower than the LRF from internal initiating events at Power (presented in Section 25.7). Contribution form the turbine missile impacting Hx/B is the highest (39 percent) because the loss of Hx/B has much higher IE frequency than any other cases. Contribution form the turbine missile impacting R/B is the second highest (37 percent). The third contributor is the turbine missile impacting B/B (22 percent). The higher contribution from the loss of the B/B to the LRF than the contribution to the CDF can be explained by the effectiveness of the systems in B/B (e.g., FLSS, containment venting) in the Level 2 PSA space: the ratio of CLRP to CCDP is highest for the loss of the B/B due to the loss of FLSS and containment venting (except for COPS).

In addition, it is qualitatively argued that the risk from turbine missiles on SFP is insignificant. Structural failure of SFP itself or spent fuels due to direct impact of turbine missile or overhead crane drop would have negligible frequency, it is prevented by design against a direct aircraft impact which would bound the impact from turbine missile. As such, un-recovered LOOP with loss of specific building/division is considered as the risk contributor. Except for Hx/B, the initiating event frequencies are in the same order or an order of magnitude lower than the LRF from internal initiating events at Power and SFP. Given multiple systems for SFP cooling/makeup are credible, the LRF of SFP from turbine missiles is deemed insignificant. The initiating event frequency for the turbine missile impacting Hx/B is two orders of magnitude higher than the LRF from internal initiating events at Power and SFP. However, at least one FPC train, two FLSS divisions and FLSR are available, which would provide a CLRP less than 1E-4 and thus make the LRF insignificant. In addition, large time available to SFP uncovery (given a LOOP event) would make the successful recoveries (i.e., repair of randomly failed components and injection) more likely than the case of reactor at Power. Overall, the risk from turbine missiles on SFP is deemed insignificant compared to the risk of reactor from internal events at Power and risk of SFP from internal events.

(6) Key Assumptions and Study Limitations

Assumptions in the turbine missiles assessment were made in the development phase. These relate to each aspects of PSA, such as the identification of initiating events, the definition of success criteria, reliability data selection, containment performance, accident progression and so on. In addition, when the level of plant requirements specification limits the detail of the PSA, then appropriate assumptions were developed to allow completion of the PSA.

Among assumptions in turbine missiles assessment, key assumptions, which have comparatively large impact on the result, have been listed from the result of sensitivity analysis.

The key assumptions considered in the turbine missiles assessment are listed below.

- Offsite power does not recover given a tornado missile induced Loss of Offsite Power (LOOP).
- Impact of turbine missile on the Reactor Building (R/B), C/B or Heat Exchanger Building (Hx/B) is limited to one safety division.
- Given a building or division is impacted by turbine missile, all the active and passive SSCs in the building or division lose the functions.
- Spurious events are not induced by turbine missile (random spurious events are considered).
- The same HEPs as used for the IEAP Level 1 PSA and Level 2 PSA are applicable to the turbine missile risk analysis (except for loss of MCR case) for the operator actions performed in the MCR. This is because the MCR is not impacted.
- The HEPs (credited in this study) for the long term actions performed at the B/B control room are not affected by a turbine missile.
- The HEPs (credited in this study) performed inside C/B but outside the MCR are not impacted by a tornado missile, except for a turbine missile impacting C/B.
- FLSR (Mobile Injection Facility) Unavailability is not impacted by a turbine missile because FLSR is operated in yard with the time available of 8 hours or more.
- The impact on the cutsets which are below 1.0E-14 /y is not captured. This is the limitation of this simplified approach.

25.11 PSA for External Hazards

The methodology described in this chapter addresses the development of the PSA for the external hazards for the UK ABWR. The external hazards PSA provides an integrated, structured analysis that assesses plant risk and identifies potential plant vulnerabilities.

This chapter describes the Level 1 and Level 2 PSA approach and results associated with the external hazards analysis for the UK ABWR PSA during the GDA phase. External hazards that are considered to be site-specific were outside the scope of the analysis and will be deferred until the pots GDA phase.

All potential external hazards were identified, and candidate external hazards were prioritised to determine which were required to be assessed further. The candidates were divided into hazards assessed during the GDA stage and those to be assessed during the post GDA phase. The methodology and the results of prioritisation are described in Section 25.11.1.

Procedures to assess each external hazard vary according to the type of external hazard. The assessment results for those external hazards assessed during the GDA are provided in Section 25.11.2.

25.11.1 Basic Approach on PSA for External Hazards

In this section, the approach for performing the prioritisation of external hazards for detailed analysis in the UK ABWR PSA for the GDA is described. The qualitative and quantitative screening analyses performed in order to identify the list of hazards to be assessed as part of external hazards PSA is provided.

25.11.1.1 Definition of External Hazard

External hazards are defined as events which pose a hazard to a site that originate from outside both the site and its processes, where the Duty Holder has no control over the initiating event. External hazards are considered in the following groups [Ref 25-93]:

- Natural hazards – those hazards that take place at the site as a result of the geophysical location and meteorological conditions, e.g., flooding, extreme wind, or seismic activity.
- Man-made hazards – those hazards that may affect a plant as a result of human presence or utilisation of an area near or adjacent to the site, e.g., external explosions, external fires, or aircraft impacts. Terrorist or other malicious acts are assessed as part of the security assessment and are excluded from this assessment.

25.11.1.2 Technical Approach and Steps for Prioritisation of Hazards

The external hazards assessment was performed consistent with the expectations of the IAEA GSR Part 4 [Ref 25-92], and IAEA SSG-3 [Ref 25-93]. A successive screening process as recommended in IAEA SSG-3 [Ref 25-93] is followed to minimise the emphasis on internal and external hazards whose contribution to risk is low and to focus the analysis on hazards that are potentially risk significant.

A consistent approach was applied to the identification of hazards for deterministic analysis and PSA, and the analysis of their contribution to core damage frequency.

The main stages of the hazards prioritisation typically include (See IAEA SSG-3 [Ref 25-93]):

(1) Collection of initial information on external hazards,

(2) Hazard identification, including single and combined hazards,

(3) Hazard screening analysis, both qualitative and quantitative,

(4) Bounding assessment, and

(5) Detailed analysis and PSA modelling.

The screening approach defined in SSG-3 [Ref 25-93] describes three processes. The first process is where items can be screened out (removed from the analysis) or screened in (retained in the analysis). For this first step, items are normally screened out qualitatively, where the hazards are not physically or realistically possible. The second process is where the hazard is grouped with or bounded by another hazard. The third process is where a hazard is carried forward to be assessed in more detail given the hazard is likely to have a significant contribution to CDF. Those hazards not selected are retained at the frequency at which they are screened. This quantitative screening process is referred to as the prioritisation of hazards.

25.11.1.3 Collection of Initial Information on External Hazards

As the starting point for the external hazards PSA, all available information specifically related to the internal and external hazards was collected. The following information was used:

(a) Design information relating to internal and external hazards as considered in the safety analysis report. Identification of (defensive) measures taken during design of the plant against hazards:
- Resistance/strength of buildings against fires, explosions, blasts, floods etc., protection barriers.

(b) Layout of plant buildings, structures, systems and components.
(c) Plant layout and topography of the proposed site and surroundings (for hazards to be assessed for the post GDA phase).
(d) Information on the location of pipelines, transportation routes and on-site and offsite storage facilities for hazardous materials.
(e) Location of industrial facilities in the vicinity of the site for hazards to be assessed for the post GDA phase.
(f) Historical information on the occurrence of any external hazards for various regions in the UK to support the GDA, or at the site for hazards to be assessed for the post GDA phase.

25.11.1.4 Identification of External Hazards

(1) Individual Hazards

The potential external hazards that could impact the plant to support the Deterministic (Design Basis) Fault Studies were identified as a comprehensive list of generic external hazards for the UK.

A necessary part of the safety case was the definition and formal statement of the generic site envelope parameters for the GDA, and site-specific parameters during the post GDA phase as well as the load capacity margins on the different nuclear safety components. The approach was further extended in [Ref 25 -163]:

- The methodology for the classification of the comprehensive list of external hazards forming the basis of the generic and site-specific external hazard characterisations.
- The proposed methodology to identify site-specific external hazards.
- Established an accepted screening criteria and process to identify the comprehensive list of external hazards for the GDA and suitable nuclear site(s), i.e., the Wylfa and Oldbury sites.
- Identified hazards that are likely to be dominant contributors determining their overall impact on the 'risk' assessment.
- Provided justification for the list of external hazards to be considered for the deterministic DBA. The decision on whether hazards have been included or excluded is based on the specified criteria and seeks to validate both its completeness and the process used to generate it, when compared to international guidance documentation.

The methodology used for the grouping and (qualitative) screening of the list of external hazards in [Ref 25 11-4] was performed in the following steps:

(1) Identification of Hazards
(2) Grouping by Denomination
(3) Grouping by Plant Effect
(4) Hazard Screening
(5) Hazard Classification (GDA or post GDA phase)

These steps are discussed further in the following sections.

(a) Identification of External Hazards

The output from this section provided a comprehensive list of potential external hazards that forms the basis of the generic and site-specific external hazard characterisations and is also sufficient to support the PSA. The list of hazards was derived from the sources listed in Table 25.11.1-1.

A comprehensive list of external hazards was derived from 11 reference sources listed in Table 25.11.1-1. This list was developed and presented in Appendix A of Topic report on External Hazard Protection [Ref 25-105] and resulted in the identification of 167 external hazards.

Table 25.11.1-1 Reference Source for Identifying External Hazard

No.	Reference Source
1	USNRC, "PRA Procedures Guide" (NUREG-CR-2300)
2	Pre-Construction Safety Report (Sizewell B PCSR)
3	OECD Nuclear Energy Agency (NEA), "Probabilistic Safety Analysis (PSA) of the Other External Events Than Earthquake"
4	WENRA RHWG, "Report Safety of new NPP designs - Study by Reactor Harmonization Working Group RHWG", March 2013.
5	USNRC, "PRA Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," & "Evaluation of External Hazards to Nuclear Power Plants in the United States," (NUREG 1407&NUREG/CR-5042)
6	IAEA, "External Events Excluding Earthquakes in the Design of Nuclear Power Plants" (IAEA Safety Guide, NS-G-1.5)
7	European Utility Requirements (EUR), Volume 2, Section 2.4, "Generic Nuclear Island Requirements: Design Basis"
8	Swedish Nuclear Power Inspectorate (SKI), "Guidance for External Events Analysis", SKI Report 02:27, Michael Knochenhauer, Pekka Louko, February 2003
9	HSE, "Technical Assessment Guide" T/AST/013, 2009
10	HSE, "Generic Design Assessment Guidance to Requesting Parties" (ONR-GDA-GD-001 Revision 0)
11	NNB Gen Co LTD, "Hinkley Point C PCSR – Redacted Sub Chapter 13.1 – External Hazard Protection – Part of chapter 13, Hazards Protection," (HPC-NNBOSL-U0-000-RET-000046 Issue 2), 22/08/2012.

(b) Grouping and Screening Criteria
An assessment of the external hazards was performed in [Ref 25-105], detailing how the original 167 external hazards were grouped and screened.
The following criteria were used in this assessment [Ref 25-105].

I. Hazard Grouping by Denomination

These are hazards that are bounded within the definition of other hazards. Grouping these hazards together eliminates hazard repetitiveness and captures a single point hazard.

II. Hazard Grouping by Plant Effect

Hazards can be grouped with other hazards based on the effect on the plant. If the hazard's potential impacts for plant are similar, these hazards are grouped to avoid repeated description of definition and protection. This process does not eliminate the hazard; instead these are grouped for rational explanation of protection policy.

III. Screening by Frequency of Occurrence

These are defined as having a frequency of occurrence less than 1E-7 /y.

(c) Identified External Hazards

After completion of the above identification, grouping and screening process [Ref 25-105], twenty-two hazards presented in Table 25.11.1-2 were identified and characterised for assessment in the PSA. Further qualitative and quantitative screening has been conducted for these identified hazards.

Table 25.11.1-2 List of Identified External Hazards

Hazard ID	Hazard Group*	Hazard ID	Hazard Group
PSA EH1	High Air Temperature	PSA EH12	External Fire
PSA EH2	Low Air Temperature	PSA EH13	Toxic Hazards
PSA EH3	High Wind	PSA EH14	External Missile
PSA EH4	Rainfall	PSA EH15	External Explosion
PSA EH5	Drought	PSA EH16	Extra-terrestrial Object (i.e. Meteorite)
PSA EH6	Snow	PSA EH17	Sand storms or salt storms
PSA EH7	Electromagnetic Interference (EMI)	PSA EH18	Industrial Environment (Radioactive release)
PSA EH8	High Sea or River Water Temp.	PSA EH19	Water based Biological Fouling
PSA EH9	External Flooding	PSA EH20	Transportation accidents (direct impact)
PSA EH10	Seismic Activity	PSA EH21	Land & Air-based Biological Fouling
PSA EH11	Aircraft Impact	PSA EH22	Flotsam/ Jetsam/Log jam

*: "hazard group" used in this section is equivalent to the term "fault group" used in the other sections.

(2) Combination Hazards Consideration

The possible combinations of hazards were identified on the basis of the list of individual hazards. The entire list of potential hazards was used for this purpose before any screening analysis was carried out. Usually, combined hazards involve only natural (thus external) hazards. However, combinations of natural hazards and man-made hazards are also possible and cannot be excluded.

The general approach used for the identification of a realistic set of combinations of hazards was based on a systematic check of the dependencies between all external hazards. In principle, the following mechanisms for combinations of hazards are considered (see IAEA SSG-3 [Ref 25-93]):

There are three distinct mechanisms in which multiple design basis external hazards may occur in combination with each other:

i. **Consequential External Hazards**

An event causing a primary hazard may give rise to one or more consequential, secondary hazards due to a direct causal relationship between the primary and secondary hazard(s).

ii. **Correlated External Hazards**
Multiple external hazards could occur as a consequence of a single underlying cause, in which case they can be assumed to be correlated. The underlying cause could be either internal or external.

iii. **Independent External Hazards**
External hazards are considered to be independent if they could only be expected to occur together by random coincidence, due to there being no causal association between the initiating events.

Correlated load combinations are dependent on the site characteristics (e.g., presence of upstream dam for a possible combination of seismic hazard and external flooding); they cannot be comprehensively assessed and have been screened out in the GDA. The analyses of hazard combinations are however important for GDA in order to identify, at an early stage, potentially significant hazard combinations which may not have been accounted for in the design. Therefore, a preliminary assessment of combination hazards was performed following the qualitative screening of the hazards for further assessment at the post GDA phase [Ref 25-106].

25.11.1.5 Qualitative Screening for Individual Hazards
The external hazards to be considered in the PSA are presented in Subsection 25.11.1.4 (1). It is necessary to review all the hazards identified in the deterministic hazards analysis and thus perform the screening analysis based on the objectives of the PSA.

(1) Screening Criteria

Screening and prioritisation criteria have been developed for the deterministic external hazards identification and the screening process based on the list of hazards from EPRI, IAEA, ONR TAGs and the SKI report. The criteria were reviewed for application to the PSA. The qualitative and quantitative screening criteria ensure that no potentially significant risk contributors from any external hazard relevant to the plant and the site are omitted for application to UK ABWR.

(2) Screening Results

The qualitative screening was conducted [Ref 25-106] under the criteria and the identified external hazards were categorised into three categories, as presented below.

Screened out:

PSA EH1 High Air Temperature
PSA EH4 Rainfall - Extreme and Intense precipitation
PSA EH5 Drought
PSA EH8 High Sea or River Water Temperature

Being deferred to the post GDA phase**:**

PSA EH7 Electro Magnetic Interference
PSA EH9 External Flooding
PSA EH12 External Fire - Pool fire and fire ball
PSA EH13 Toxic Hazard
PSA EH14 External Missile

PSA EH15 External Explosion

PSA EH17 Cloud/Storms (Ash, Dust, Sand, Salt)

PSA EH18 Industrial Environment

PSA EH19 Water based Biological Fouling

PSA EH20 External Transport Impacts

PSA EH21 Land & Air-based Biological fouling

PSA EH22 Flotsam/Jetsam/Log Jam

Retained and being assessed in the quantitative prioritisation:

PSA EH2 Low Air Temperature (Deferred to SLA for Class 3 HVAC)

PSA EH3 Wind, (Wind-blown debris, Tornado)

PSA EH6 Snow

PSA EH10 Seismic Activity - Ground Motion, Liquefaction

PSA EH11 Aircraft Impact

PSA EH16 Extra Terrestrial Object

25.11.1.6 Identification and Screening of Combination Hazards for EH PSA

The analyses of hazard combinations are also important for the GDA in order to identify, at an early stage, potentially significant hazard combinations which may not have been accounted for in the design. Therefore, the hazard combinations were assessed with the following caveats and assumptions.

(1) Hazard combinations where there is strong dependency on-site selection have been deferred to the post GDA phase.
(2) Hazard combinations which are considered to be independent are not assessed due to the low frequency of coincidental hazards above the design basis occurring [Ref 25-106].
(3) Combinations that contain hazards that have been screened out qualitatively and that are considered not to have an additive effect (i.e., the impacts of the hazards do not add) on either hazard in the combination, are also screened out.
(4) Combination hazards that are considered to be relevant for the GDA phase are identified and listed.

The list of non-independent hazards (i.e., the consequential and correlated combination hazards [Ref 25-165]) were reviewed using a pairwise comparison method [Ref 25-166] and specific combination hazards identified to be assessed in the PSA. In addition, external hazards as defined [Ref 25-106] were then reviewed in order to ensure that there are no additional combination hazards that needed to be analysed. The final sets of nine PSA combination hazards identified for further assessment are listed in Table 25.11.1-3.

The combinations of external hazards with internal hazards will need to be assessed once the assessment of the internal hazards and external hazards has been completed. This will allow realistic combinations to be assessed, based on detailed understanding of the potential impacts from the external hazards and where internal hazards pose a risk. For example, although both the individual hazards included in PSA CH 3 (External flooding (Rain) + Snow) are screened out, the potential load combinations were not assessed in the design basis and there is insufficient structural analysis from the hazards safety case to determine the capacity of the structures within scope of the GDA to withstand the loads for this combination hazard. Due to one or more of the individual hazards in the remaining combination hazards above being identified as retained for further analysis or deferred to the post GDA phase, a more detailed analysis of the hazard combinations is not possible at this time.

Table 25.11.1-3 Combination Hazards Description

PSA CH ID	Combinations	CW Building / HX Building	Switchyard + Offsite Power	DG Building	Backup Building
PSA CH 1	High Wind + External Flooding	Loss of Ultimate Heat Sink (LUHS) (beyond DBA Events) due to External Flooding	LOOP due to High Winds	Loss of cooling to DGs (Loss of RWS)	
PSA CH 2	Low Air Temperature + Snow	LUHS due to ice formation in CW building. Snow loads on buildings	LOOP due to Ice formation on conductors, (Rime ice)	Snow loads on buildings	Snow loads on buildings
PSA CH 3	External Flooding (Rain) + Snow	LUHS due to flooding in CW building. On-site flooding affecting operations Rain-on-Snow loads on buildings		Rain-on-Snow loads on buildings	Rain-on-Snow loads on buildings
PSA CH 4	Low Temperature + High Winds	LUHS (beyond DBA Events) Freezing of Water – Ice floes / Frazil ice	LOOP due to High Winds	Low Temperature affecting DG Fuel, Connecting Tunnels Loss of cooling to DGs	Low Temperature affecting DG Fuel, Connecting Tunnels
PSA CH 5	Seismic + External Flooding	External flooding due to tsunami from seismic event / flooding debris	LOOP due to Seismic Events	Loss of cooling to DGs (loss of RSW)	
PSA CH 6	Low Temperature + Rain (Ice Storms)	LUHS due to icing of CW building	LOOP due to ice storms	Low Temperature affecting DG Fuel, Connecting Tunnels Loss of cooling to DGs	Low Temperature affecting DG Fuel, Connecting Tunnels
PSA CH 7	Snow + High Winds (Snow Drifting)	LUHS (beyond DBA Events) Freezing of Water – Ice floes / Frazil ice	LOOP due to High Winds	Low Temperature affecting DG Fuel, Connecting Tunnels Loss of cooling to DGs	Low Temperature affecting DG Fuel, Connecting Tunnels
PSA CH 8 (post GDA phase)	Water Based Biological Fouling + High Seawater Temp. / High Air temp.	Reduced effectiveness of UHS and HVAC		Reduced effectiveness of UHS and HVAC	
PSA CH 9 (post GDA phase)	High Wind + External Transport Accident (Shipping Accident)	Loss of UHS due to shipping accident in CW building	LOOP due to High Winds	Loss of cooling to DGs	

25.11.1.7 UK ABWR Building Screening Based on Safety Significant Equipment

(1) UK ABWR Buildings in the GDA Scope

The standard UK ABWR plant arrangement consists of the following main buildings:

- Reactor Building (R/B),
- Control Building (C/B),
- Heat Exchanger Building (Hx/B),
- Turbine Building (T/B),
- Radwaste Building (Rw/B),
- Service Building (S/B),
- Backup Building (B/B),and
- Emergency Diesel Generator Building (EDG/B).

The standard plant is located on a site adjacent or close to a body of water with sufficient capacity to provide cooling under all operating conditions.

The C/B that houses the main control room is located at the central location as the hub of the plant operating staff's activities. The T/B is oriented in such a way, that any plane perpendicular to the turbine generator axis does not intersect with the R/B and C/B in order to minimise the probability of a turbine missile striking any Safety Related Structures, Systems, and Components (SSCs). This standard UK ABWR plant arrangement is shown in the PCSR. The items not in the GDA scope are the Circulating Water Structure and the Spent Fuel Storage Facility, which leaves 26 items to be included as part of the qualitative screening process.

For GDA purposes, some plant equipment was not included, e.g., the switchyard, and other non-essential equipment in the yard. In addition, some of the measures credited to protect against severe accidents are not fixed facilities in the backup building and were not explicitly considered in the assessment of buildings for hazards prioritisation. These measures were used to mitigate against severe accident sequences where multiple safety facilities have been lost (e.g., earthquake, flooding, aircraft crash, tornado, etc.).

As these systems are specifically designed to mitigate the severe accident (beyond design basis) conditions, they can be credited for the mitigation of external hazards. These mobile facilities were not credited for the short-term mitigation strategies and they were assumed to be secure and remain accessible given the external hazard.

(2) Screening Results for Further Analysis

In the first step of the screening process, the buildings on the site within the GDA scope were reviewed to identify the buildings that are important to the safety of the facility. The following criteria were used to determine whether a building should not be screened out from the assessment:

- The hazard in a building causes an IE,
- The hazard in a building affects SSCs needed for accident mitigation or the building contains safety-related SSCs, and
- The hazard in a building causes the above criterion 1 or 2 in a neighbouring building, when its effects are not contained in the building.

The SSCs required to safely shutdown the plant following a hazard were identified from the internal events PSA models. All SSCs that are credited to mitigate an initiating event have been identified. In addition to the SSCs identified in the buildings above, SSCs in the yard or in the switchyard have also been identified.

The Service Building, Suppression Pool Tank and Suppression Pool Tank Service Tunnel were qualitatively screened. Building collapsing onto the Class 1 buildings was not considered a realistic failure mode because the design basis of the buildings adjacent to the Class 1 buildings will take into account the need to protect these buildings against structural collapsing. The buildings retained for further analyses are listed in Table 25.11.1-4.

Table 25.11.1-4 UK ABWR Buildings Retained for Further Analyses

Building & Facility No.	Object name	Class / Category
101	Reactor Building	A-1
102	Control Building	A-1
103	Heat Exchanger Building	A-1
104	Radwaste Building	B-2
106	Main Stack	A-1
107	Back-up Building	A-2
108	Turbine Building	B-2
110	Emergency Diesel Generator Building (x3)	A-1
505	Condensate Storage Tank	A-2 for 513 not listed
515	Water Storage Tank for B/B (x2)	A-2
601	R/B - EDG/B Connecting Service Tunnel (x3)	A-1
502	Light Oil Storage tank	A-2
602	B/B - LOT Connecting Service Tunnel	A-2
603	RCW Tunnel	A-1
604	R/B - B/B Connecting Service Tunnel	A-2
605	R/B - CST Connecting Service Tunnel	A-2
618	B/B - Water Storage Tank Connecting Service Tunnel (x2)	A-2

25.11.1.8 Quantitative Screening for EH PSA
(1) Screening Results and a Brief Summary

A list of external hazards to be considered in the PSA has been identified in Subsection 25.11.1.4 and the qualitative screening was conducted in Subsection 25.11.1.5 (2).

Through the review of hazards, the External Flooding, Toxic Release, External Explosion, Water Based Biological Fouling, and External Transport Hazards have been deferred to the post GDA phase [Ref 25-106]. **Seismic** activities were assessed independently; the Seismic PSA is described in Section 25.11.2.

The quantitative prioritisation of the **Low Air Temperature Hazard**, **Extreme Winds Hazard**, **Snow Hazard**, **Aircraft Impact** and **Extra Terrestrial hazard** is summarised in 25.11.1.8 below.

(2) The Quantitative Prioritisation of External Hazard

(a) PSA EH2 Low Air Temperature Hazard

The Class 1 HVAC design parameters have a significant margin above the best estimate values that have been assessed for Wylfa. For example, the 1E-8 best estimate value of -12.2 ^{o}C is significantly less severe than the UK ABWR GDA 1E-4 design basis value of -22.5 ^{o}C [Ref 25-106].

Low air temperature hazard affecting Class 1, Class 2 and Class 3 HVAC systems is expected to affect the internal environment of the buildings and affect temperature sensitive equipment. The HVAC systems including Class 3 are expected to continue operating beyond external temperature design parameters and sufficient margin is demonstrated that the internal temperatures will not lead to failure of temperature sensitive equipment. This margin is even greater when considering best estimate hazard curves for the low temperature hazard. It is expected that Hitachi-GE will be able to provide the justification for continued operation of the normal Class 3 HVAC systems below the design operating temperatures during the post GDA phase.

The low air temperature hazard is therefore not normally expected to result in a plant trip (manual or automatic) or require a controlled manual shutdown due to plant limits and conditions so the hazard was screened out. The possible LOOP consequence is already included in the weather-related LOOP frequency.

The risk from the frazil ice and sea ice in the CW intake buildings is dependent on the CW building design and layout of the RSW pit in the CW building. It is also dependent on the likelihood that sufficient water can be passed through the drumscreens to the RSW pits to supply the demand from the RSW pumps. The cooling water building was out of scope for the GDA, so hazards affecting the CW building were not assessed for the GDA. Therefore, frazil ice and sea ice in the CW intake building is therefore deferred to the post GDA phase.

Rime ice was screened out on the basis that this hazard has already been included in the weather-related LOOP > 14 hours initiating event.

(b) PSA EH3 Extreme Winds Hazard

High Wind

The seismic capacity of Seismic Category 2 buildings (Class 2) is approximately 50 percent of that for Seismic Category 1 buildings. The Wind Load has to be at least ~50 times greater than the Design Wind Load to exceed the Design Seismic Load, [Ref 25-106] which means the velocity of the wind will need to be approximately 5 times higher than that of a Class 1 building design Wind Load (43.4 m/s), i.e., over 217 m/s. Therefore, there is a significant margin to sustain high winds at a return frequency significantly less than 1E-4 /y for Category 1 and Category 2 buildings.

The 1E-8 best estimate high wind frequencies were less than the 1E-4 design basis wind speeds. The structural capacities of Class 1 and Class 2 buildings need to be assessed in detail, and the findings here should be confirmed in the post GDA phase. However, the seismic design appears to provide significant margins for extreme straight wind hazards.

Tornado Missiles

The UK ABWR buildings are designed to withstand tornado missiles in compliance with NRC RG 1.76 [Ref 25-109] and NUREG-0800 [Ref 25-110].

The capacity of the LOT to withstand tornado missiles has not been determined at the current design stage. Although the LOT support will be seismically designed for the DBE, the tank itself has not been designed, and is assumed to be a steel tank that is not able to withstand the design basis tornado missiles. Assuming a DBE high wind / T3 tornado results in damage to the LOT that provides longer term fuel oil supplies to the DGs in the backup building, the station could be more vulnerable to Station blackout. For the loss of the backup building given LOOP, the bounding CDF of this hazard did not meet the screening criteria and was therefore retained for further analyses. Ensuring diverse and adequate power and cooling supplies for those hazards with the potential of losing the long term offsite power supplies is essential and the structural design of the LOT can be seen here to be an important element in maintaining the long term offsite power. It would therefore be expected that the LOT be designed to withstand all design basis hazards at a return frequency of 1E-4 /y (deterministic), which is to at least withstand the tornado missiles that could be generated with a wind speed of 42 m/s.

It should be noted that as tornadoes are most likely to occur in the south and east of England, tornados are much less likely at the sites chosen for the UK ABWR, i.e., Wylfa and Oldbury, so the frequencies of tornadoes at these sites would need to be developed.

(c) PSA EH6 Snow Hazard

The structural capacities of the buildings (seismic structural load and wind loads) bounded the maximum credible snow loads for the UK with a considerable margin. The snow hazard was therefore be screened out in the PSA of UK ABWR based on the low frequency of occurrence and the built-in capacity of the safety related buildings, as the event is of equal or lesser damage potential than similar events for which the plant has been designed.

Snow hoods are installed to protect HVAC systems from snow. Protection of HVAC systems against high levels of snow should be confirmed during the detailed design of the HVAC systems. Load combinations such as rain-on-snow should also be evaluated and confirmation provided that sufficient margin exists for the combination hazards.

(d) PSA EH11 Accidental Aircraft Impact

The design of the reactor building provides sufficient protection against accidental aircraft impact to protect the primary containment and spent fuel pool. Damage to any one of the three divisions in the R/B is also unlikely and would only be susceptible to a direct impact from the largest aircraft. The risk from Aircraft Impact Assessment (AIA) on the reactor building was low and the frequency of an impact on the R/B would be below the screening criteria [Ref 25-106].

The control building is mostly protected by other buildings on four sides (Reactor Building, Turbine Building, Service Building and the Radwaste Building). Direct impact with the control building could however degrade the operations from the Main Control Room (MCR) and affect multiple divisions of the reactor protection system. The turbine building, heat exchanger building, cooling water inlet building and radioactive waste building were all assumed to be failed by the aircraft impact hazard and did not meet the screening criteria and therefore need to be assessed in the PSA.

Although an aircraft impact might fail these buildings, the normal RHR shutdown systems and ECCS systems in the reactor building were unaffected and therefore safe shutdown was performed using the normal shutdown systems and was not significant compared to the other AIA events. AIA for these structures were screened from further assessment in the PSA.

Although some buildings could be screened out from the AIA hazard, insufficient information is available to fully assess the capacities of the buildings to withstand accidental aircraft impacts, e.g. control building. The frequency of an aircraft crash affecting various buildings and the potential risk from SSC failures as a result of the AIA, were such that the quantitative screening criteria could not be fully applied to all buildings, and this hazard is therefore retained for further assessment in the PSA.

(e) PSA EH16 Extra Terrestrial Object

The impact of the most frequent damaging meteoroid in all cases, except for a direct strike on the roof of the MCR, resulted in potential loss of a single train or division. The highest CCDP for a meteorite affecting a single division or train (without simultaneous LOOP) is based on a General Transient w/o operator actions within 1 hour or with operator actions only possible in MCR set as failed (CCDP = 4.2E-05) [Ref 25-106]. The combinations of the low frequency of the larger meteorite strikes, resilience of the buildings against the small meteorite strikes and the availability of alternative safe shutdown paths were sufficient to demonstrate that the hazard is not risk significant (risk is below 1E-08 /y).

Following this bounding assessment, the CDF for all combinations of potential impacts of meteorites is less than 1E-8 /y. The hazard was therefore quantitatively screened out based on the criteria for small meteorite strikes (the criteria is where hazard frequency < 1E-7 py and less than 1 percent of the internal events CDF.

25.11.1.9 Results of Prioritisation of External Hazards PSA

The external hazards identified for PSA were reviewed against the qualitative screening criteria and the results of the screening analysis are presented in Subsection 25.11.1.5.

A summary of the results of the screening analysis on the buildings or facilities for external hazards screened in (excluding the seismic hazard) are presented in Table 25.11.1-5 below. A summary of the results of the screening and prioritisation analysis of the external hazards not deferred to the post GDA phase is presented in Table 25.11.1-6. The maximum plant effect from a hazard, the representative initiating event, the SSCs or systems required to mitigate the hazard, and the results of the prioritisation or screening analysis are provided identifying whether the hazard was screened qualitatively, deferred to the post GDA phase or retained for further analysis. Following more detailed and bounding assessments, the snow hazard and extra-terrestrial object hazards were quantitatively screened. Following the qualitative and quantitative screening process, the following hazards were screened out for the GDA:

PSA EH1	High Air Temperature
PSA EH2	Low Air Temperature
PSA EH4	Rainfall - Extreme and Intense precipitation
PSA EH5	Drought
PSA EH6	Snow
PSA EH7	Electro Magnetic Interference
PSA EH8	High Sea or River Water Temperature
PSA EH9	External Flooding
PSA EH12	External Fire - Pool fire and fire ball
PSA EH13	Toxic Hazard
PSA EH14	External Missile
PSA EH15	External Explosion
PSA EH16	Extra Terrestrial Object
PSA EH17	Cloud/Storms (Ash, Dust, Sand, Salt)
PSA EH18	Industrial Environment
PSA EH19	Water based Biological Fouling
PSA EH20	External Transport Impacts
PSA EH21	Land & Air-based Biological fouling
PSA EH22	Flotsam/Jetsam/Log Jam

Following the prioritisation assessment, the following hazards were retained for further analysis in the PSA [Ref 25-106]:

(1) PSA EH3 Extreme wind hazard due to potential impact of wind borne missiles on LOT (Bounding CDF ~2.2E-07 /y)

(2) PSA EH10 Seismic Activity - Ground Motion, Liquefaction

(3) PSA EH11 Aircraft Impact on control building leading to loss of MCR and loss of safety divisions (CDF to 7.85E-08 /y), aircraft impact on the turbine building (CDF to 1.1E-08 /y) and aircraft impact on the Heat Exchanger Building (CDF to 1.4E-08 /y). Aircraft impact on CW building impact will also need to be assessed (no frequency provided). The frequency of an aircraft impact on the Radwaste building is calculated as 6.57E-07 /y. No risk assessment has been developed for the Radwaste facility for GDA at this point in time and the risk from aircraft impact on this facility will need to be considered in the Radwaste facility risk assessment.

Table 25.11.1-5 Results of Screening and Prioritisation Analysis for Buildings (1/2)

Building & Facility No.	Building	High Air Temp. (EH1)	Rainfall (EH4)	Drought (EH5)	High Sea Water Temp.(EH8)	Low Air Temp. (EH2)	Accidental Aircraft Impact	Wind (EH3)	Snow (EH3)	Comments
101	Reactor Building	SC 6	SC 6	SC 6	SC 6	SC 1	SC12	SC 1	SC1 Snow Load	Cat A, Class 1 building
102	Control Building	SC 6	SC 6	SC 6	SC 6	SC 1	Not screened	SC 1	SC1 Snow Load	Cat A, Class 1 building
103	Heat Exchanger Building	SC 6	SC 6	SC 6	SC 6	SC 1 for Class 1 HVAC, Class 3 HVAC screened out for GDA	SC12	SC 1	SC1 Snow Load	Cat A, Class 1 building
104	Radwaste Building	Not Assessed	Not Assessed	Not Assessed	Not Assessed	Not Assessed	Not Assessed	Not Assessed	Not Assessed	Release of radioactivity from Radwaste Facility needs to be assessed.
106	Main Stack	No impact on SSC	No impact on SSC	No impact on SSC	No impact on SSC	No impact on SSC	Main stack failed (on impact with RB, no impact on vent function -)	No impact on SSC	No impact on SSC	Manual shutdown expected
107	Back-up Building	SC 6	SC 6	SC 6	SC 6	SC 1	SC12	SC 1	SC1 Snow Load	Cat A, Class2 building
108	Turbine Building	SC 6	SC 6	SC 6	SC 6	Class 3 HVAC screened out for GDA	SC13	SC 1	SC1 Snow Load	Cat 3 Switchgear rooms may exceed design temperature for temp. sensitive equip.
109	Service Building	Building screened	Building screened	Building screened	Building screened	Building screened	Building screened	Building screened	Building screened	S/B is adjoining to East side of Control Building Based on review of Layout of these two buildings and the civil design characteristics this building could be screened. It is expected any potential for significant consequential hazards will be identified in these assessments (e.g. Seismic-fire interactions).
110	Emergency Diesel Generator Building	SC 6	SC 6	SC 6	SC 6	SC 1	SC12 (as per B/B)	SC 1	SC1 Snow Load	Loss of 2 DGs is Tec Spec admin shutdown condition - Not screened -assessed for External Hazards as hazard impacting one building likely to impact a second building.
502	Light Oil Storage Tank, Foundation	No impact on SSC	No impact on SSC	No impact on SSC	No impact on SSC	No impact on SSC	SC12 (as per B/B)	LOT failed due to turbine missile (Tornado)	SC1 Snow Load	No necessity for immediate manual shutdown. Light Oil storage tank could be result in a consequential internal hazard Fuel oil only required for longer term oil supplies
505	Condensate Storage Tank, Foundation	No impact on SSC	No impact on SSC	No impact on SSC	No impact on SSC	No impact on SSC	CST – not assessed (bound by B/B)	No impact on SSC	SC1 Snow Load	Flooding potential of tank should be analysed; Identified internal hazards will not affect foundation
508	Suppression Pool Tank, Foundation	Building screened	Building screened	Building screened	Building screened	Building screened	Building screened	Building screened	Building screened	S/P tank only used when suppression pool is pumped empty to inspect the liner (infrequent maintenance event)
515	Water Storage Tank for B/B	No impact on SSC	No impact on SSC	No impact on SSC	No impact on SSC	Consider potential freezing of outdoor piping	SC12 (as per B/B)	No impact on SSC	SC1 Snow Load	No requirement for shutdown (Technical Specification) Failure of tank has consequential hazard with potential for flooding. The water in the B-WT is used for FLSS, and its pumps are inside the B/B. Therefore, B-WT will be located near to the B/B as possible.

Table 25.11.1-5 Results of Screening and Prioritisation Analysis for Buildings (2/2)

Building & Facility No.	Building	High Air Temp. (EH1)	Rainfall (EH4)	Drought (EH5)	High Sea Water Temp.(EH8)	Low Air Temp. (EH2)	Accidental Aircraft Impact	Wind (EH3)	Snow (EH3)	Comments
601	R/B - EDG/B Connecting Service Tunnel (1)	No impact on SSC	No impact on SSC	No impact on SSC	No impact on SSC	No impact on SSC	Not assessed (bound by B/B	No impact on SSC	No impact on SSC	
602	B/B - LOT Connecting Service Tunnel (1)	No impact on SSC	No impact on SSC	No impact on SSC	No impact on SSC	No impact on SSC	SC12 (as per B/B)	No impact on SSC	No impact on SSC	
603	RCW Tunnel (1)	No impact on SSC	No impact on SSC	No impact on SSC	No impact on SSC	No impact on SSC	SC12 (as per Hx/B)	No impact on SSC	No impact on SSC	Since RCW related pipes and cables are arranged in R-H/C, R-H/C must be separated for countermeasure of hazard to avoid simultaneous damage. Therefore, the 3 R-H/Cs are prepared for each unit.
604	R/B - B/B Connecting Service Tunnel (1)	No impact on SSC	No impact on SSC	No impact on SSC	No impact on SSC	Consider potential freezing of outdoor piping	SC12 (as per B/B)	No impact on SSC	No impact on SSC	To connect the pipes and cables of alternative cooling system and to connect other various kind of pipes and cables
618	B/B - Water Storage Tank Connecting Service Tunnel (1)	No impact on SSC	No impact on SSC	No impact on SSC	No impact on SSC	Consider potential freezing of outdoor piping	SC12 (as per B/B)	No impact on SSC	No impact on SSC	
608	SPT Connecting Service Tunnel (1)	Building screened	Building screened	Building screened	Building screened	Building screened	Building screened	Building screened	Building screened	To connect suppression water piping between R/B and SPT, normally isolated.
NA	Diverse Alternate Generator (DAG)	Assumed to be available following the hazard	Assumed to be available following the hazard	Assumed to be available following the hazard	Assumed to be available following the hazard	Assumed to be available following the hazard	Assumed to be available following the hazard	Assumed to be available following the hazard	Assumed to be available following the hazard	Building classification, safety class, location etc. to be determined.
NA	CW Building	Not assessed	Not assessed	Not assessed	Not assessed	Not assessed	Not assessed	Not assessed	Not assessed	Out of scope for GDA – Assess in the post GDA phase

Notes:

(1) Cabling assumed not to fail if service tunnels have been flooded (Ref: Internal Flood PSA).

(2) SC1, SC6, and SC12 are the screening criteria defined in [Ref 25-106].

(3) HVAC systems are specific per building and consist of the following sub-systems:

(a) Reactor Area (R/A) HVAC
(b) Reactor Building Emergency Electrical Equipment Zone (RBEEE/Z) HVAC
(c) Emergency D/G Electrical Equipment Zone (DGEE/Z) HVAC
(d) Reactor Internal Pump Adjustable Speed Drive Zone (RIP ASD/Z) HVAC
(e) Turbine Building (T/B) HVAC
(f) Turbine Building Normal Electrical Equipment Zone (TBNEE/Z) HVAC
(g) Heat Exchanger Building Normal and Emergency (Hx/B-N and Hx/B-E) HVAC
(h) Control Building Emergency Electrical Equipment Zone (CBEEE/Z) HVAC
(i) Control Building Class 2 Electrical Equipment Zone (CBC2EE/Z) HVAC
(j) Main Control Room (MCR) HVAC
(k) Radwaste Building (Rw/B) HVAC
(l) Radwaste Building Normal Electrical Equipment Zone (RWNEE/Z) HVAC
(m) Service Building (S/B) HVAC
(n) Backup Building Electrical Equipment Zone (BBEE/Z) HVAC
(o) Backup Building Emergency Control Room (BBECR) HVAC

Table 25.11.1-6 Results of Screening and Prioritisation Analysis

PSA External Hazard	Maximum plant effect	Representative Initiating Event	SSCs required to mitigate the hazard	Screening
PSA EH1 **High Air Temperature**	Common cause failure of CBEEE/Z HVAC, RBEEE/Z HVAC, TB HVAC leading to CCF of RPS (Temperature sensitive equipment)	Single and common cause failure of CBEEE/Z HVAC	Precautionary manual shutdown, normal shutdown systems.	Qualitative
PSA EH2 **Low Air Temperature**	Ice formation on water exposed to low temperatures, ice formation in the Ultimate Heat Sink (UHS) (Frazil, Rime) and Soil Frost. Icing of HVAC, LOOP	LOOP > 14 hours (not recoverable) due to failure of TB HVAC, or LUHS due to frazil ice.	Rx trip due to, EDGs and normal shutdown systems available. Icing of HVAC although frost coils (heating steam) will be installed to prevent freezing of bag type filters.	Retained for further analysis
PSA EH3 **High Wind**	Loss of light oil tank for B/B, LOOP	LOOP > 14 hours (not recoverable), B/B unavailable	Rx trip due to LOOP, EDGs and normal shutdown systems.	Retained for further analysis
PSA EH4 **Rainfall**	None - Maximum rainfall within design basis.	None	Precautionary flooding measures, precautionary manual shutdown, normal shutdown systems.	Qualitative
PSA EH5 **Drought**	None- Drought / max low water levels within design basis	None	Precautionary manual shutdown, normal shutdown systems.	Qualitative
PSA EH6 **Snow**	Long term LOOP (assumed not recoverable within 24 hours) - Maximum snow loads within seismic design basis, assumed to result in LOOP, HVAC intakes protected with snow hoods	LOOP > 14 hours (not recoverable)	Rx trip due to LOOP, EDGs and normal shutdown systems.	Qualitative
PSA EH8 **High Sea or River Water Temperature**	None - High temperatures within design basis, high temperature affects efficiency	None	Precautionary load reduction or manual shutdown, normal shutdown systems.	Qualitative
PSA EH10 **Seismic Activity**	See Seismic PSA (See Section 25.11.2)	See Seismic PSA (See Section 25.11.2)	See Seismic PSA (See Section 25.11.2)	See Seismic PSA (See Section 25.11.2)
PSA EH11 **Aircraft Impact**	Direct impact on C/B, loss of MCR and multiple division of RPS	CCF of RPS, no credit for operators within 1 hour	Backup building control panel + FLSS	Retained for further analysis
PSA EH16 **Extra-terrestrial Object (i.e. Meteorite)**	Direct impact on C/B, loss of MCR and multiple division of RPS	CCF of RPS, no credit for operators within 1 hour	Backup building control panel + FLSS	Quantitative screening based on SC14

25.11.1.10 Conclusion

The results and conclusions of the qualitative and quantitative screening processes are as follows:

- The external hazards were identified and characterised.
- The external hazards were either qualitative screened, deferred to the post GDA phase or identified for the GDA assessment.
- Combinations of external hazards were identified for further assessment.
- Buildings were screened on their safety significance: the GDA scope and safety implications.
- A preliminary review and quantitative screening of the external hazards retained for PSA has been provided.
- Seismic activities were assessed independently and are described in Section 25.11.2.

Table 25.11.1-5 includes the list of buildings considered to have safety significance for the PSAs.

25.11.2 Seismic PSA

25.11.2.1 Background and Objective

The purpose of this section is to summarise the UK ABWR Seismic PSA [Ref 25-114] for both the reactor and SFP during at power and shutdown operations. However, for the seismic risks associated with shutdown conditions, only a semi-quantitative evaluation was performed. “Semi-quantitative evaluation” means the application of relatively simpler methods, such as no consideration of success paths.

25.11.2.2 Probabilistic Seismic Hazard Analysis

The input hazard data for the Seismic PSA for the UK ABWR is defined [Ref 25-114]. In this seismic PSA, the CDF, Fuel Damage Frequency (FDF) and Large Release Frequency (LRF) have been calculated.

Seismic PSA quantification results are described in Section 25.11.2.6, which also includes CDF (or FDF) and LRF results for each hazard level.

25.11.2.3 Identification of the SSCs for Seismic PSA

For the seismic PSA, seismic fragilities for the individual SSCs required to safely shutdown the reactor and to maintain a stable state condition in both the reactor and SFP have been evaluated. The list of these SSCs is called the Seismic Equipment List (SEL).

The SEL was basically developed from the Internal Events At Power (IEAP) Level 1 and Level 2 PSA for the reactor and for the SFP. In the SEL, all basic events that were modelled in IEAP PSA and the fragility parameters applied to each SSC have been included.

Furthermore, additional SSCs and/or failure modes have been modelled and included in the SEL, if these SSCs could survive in an earthquake.

In the Seismic PSA model, Seismic Category 3 SSCs were not included because SSCs that are designed as Seismic Category 3 are not credited as surviving the relatively high ground motion which would likely dominate the overall seismic PSA results.

25.11.2.4 Seismic Fragility Analysis

The seismic fragility analysis has been performed for all items in the SEL. In the fragility analysis, there are several assumptions made regarding fragility calculations because the design of most equipment has not been completed at this stage. As the design progresses, these assumptions will be verified or changed to reflect the actual conditions.

Seismic fragility was calculated for SSCs, such as Building; SSCs Similar to SSCs in Existing Plants in Terms of Specification, and SSCs Not Similar to SSCs in Existing Plants in Terms of Specification.

The evaluation results of the seismic fragility are shown in [Ref 25-114] and are composed of the following parameters.

- A_m: Median peak ground acceleration corresponding to 50 percent failure probability
- β_r: The logarithmic standard deviation of the randomness which determines the curve slope
- β_u: The logarithmic standard deviation of the uncertainty which is a measure of the spread from the median curve.

- β_c: When the composite fragility curve is used, the total variability about the median value is taken to be the square root of the sum of the squares (SRSS) of the randomness and uncertainty components.

$$\beta_c = \sqrt{\beta_r{}^2 + \beta_u{}^2}$$

- A_{HCLPF}: Ground acceleration corresponding to High Confidence of Low Probability of Failure.

25.11.2.5 Seismic PSA Methodology

(1) Initiating Event Analysis

Seismically induced initiating events that give rise to significant accident sequences and/or significant accident progression sequences are identified in this section. The summary of initiating events in seismic PSA is given in following subsection.

(a) Initiating event for the reactor

The initiating-event selection process was formed in a hierarchal manner to ensure that every earthquake greater than a size which causes LOOP was developed in the Support System Event Tree (SSET). The SSET for the reactor initiating event is shown in Figure 25.11.2-1.

Support system event trees (SSET) and each transferred event tree are explained below.

Heading "SE"

This top heading denotes occurrence of an earthquake as an Initiator.

Heading "S-LOOP"

This heading denotes the access to electricity after loss of offsite power. Loss of offsite power initiated by a seismic load was conservatively assumed to occur without recovery in this seismic PSA.

Heading "SI"

This heading denotes the structures and buildings such as the reactor building. Damage to equipment contained in the buildings is likely after the structural failures. Failure of these structures and buildings was assumed to cause an unmitigated sequence and is further assumed to lead to fuel damage.

Heading "S-ELOCA"

This heading denotes the excessive LOCA (E-LOCA) such as RPV failure or multiple failure or large piping due to structural failure (e.g., failure of RPV stabiliser, RPV pedestal or reactor shield wall). This E-LOCA was assumed to result in fuel damage (it cannot be mitigated).

Heading "CI"

This heading denotes control and instrumentation functions such as the control panels, instrumentation racks, cable trays and so on. Class 1 C&I and Class 2 C&I SSCs are considered as each basic event below an "AND" gate. However, Class 1 instrumentation racks are considered below an "OR" gate because if the Class 1 instrumentation racks fail, the C&I function is completely lost. Failure of these SSCs causes the loss of multiple safety functions and is assumed to lead to fuel damage.

Heading "S-BYP"

This heading denotes the containment isolation function such as the isolation valves which are opened during at power conditions and piping outside the containment. Failure of these SSCs causes a containment bypass sequence and may damage equipment outside the containment, similar to LOCA outside containment or ISLOCA. ISLOCA is caused by human error, which is not seismically induced. Therefore, BOC was modelled in the seismic PSA.

Heading "S-LOCA"

This heading denotes the LOCA due to external leak of valves. In addition, LOCA due to failure of piping is also considered.

For the probabilistic treatment of this LOCA and bypass (BOC) due to piping failure, the generic piping fragility based on the seismic PSA implementation guideline prepared by EPRI has been used with weighting factors based on the piping length of each.

Heading "S-F-CRE"

This heading denotes the control rod insertion functions such as HCU (Hydraulic Control Unit) and related SSCs. This heading also includes structural components (reactor internals) which maintain the geometrical conditions for rod insertion. Failure at this branch indicates that control rod insertion has failed because of failure of the CR drive function or excess deformation against insertion of the rods. If rod insertion fails, this sequence is transferred to the STE4-A event tree which is the same as the internal events ATWS with long term LOOP condition.

Heading "S-DC1"

This heading denotes Class 1 DC electrical power. If it fails, both Class 1 DC and AC power are unavailable because the EDGs cannot be initiated under LOOP.

Heading "S-AC1"

This heading denotes Class 1 AC electrical power. If it fails, Class 1 AC power is not available because of failure of the EDGs or electrical power supply systems. This is the same heading used in the seismic reactor PSA at Power.

For the shutdown seismic PSA, heading "S-F-CRE" was deleted because it is already established. Heading "S-DC1" was also deleted because it does not cause any initiating event during shutdown conditions.

The SSET end points are defined as OK (success) or core damage sequences, or are transferred to other event trees. The SSET end points are summarised as follows.

SE01	:	Success sequence
STE4-LOOP	:	Transferred to "STE4-LOOP" ET to address the LOOP event
STE4-SBO	:	Transferred to "STE4-SBO" ET to address the LOOP with loss of Class 1 AC power event
STE4-DC1	:	Transferred to "STE4-DC1" ET to address the LOOP with loss of Class 1 DC power event
STE4-A	:	Transferred to "STE4-A" ET to address the LOOP with ATWS (Anticipated Transient Without Scram) event
S-AS1S2	:	Transferred to LOCA inside containment ETs.

BYP	:	Transferred to BOC ETs.
SE08	:	Core damage sequence.
S-S4	:	Core damage sequence but transferred to "S-S4" ET to allocate appropriate PDS.
SE10	:	Core damage sequence.

Direct Reactor challenge

In the internal events PSA, initiating events caused by reactor related accidents, referred to as "reactor challenges", are also considered. In the seismic PSA, these reactor challenges were treated as described below.

ISLOCA/BOC: These were considered as seismically induced initiating events in the reactor related seismic PSA. ISLOCA/BOC could simultaneously occur with other seismic effects.

Containment failure effects such as steaming, hydrogen and deformation: These effects have been considered in the Containment Event Tree (CET) in the reactor PSA, which were derived as the "end point" of the accident sequences. However, for the UK ABWR seismic PSA, the starting point of these are LOOP in the seismic PSA. The LOOP impact as an initiating event on the SFP has already been considered in the timing of the earthquake.

(b) Initiating event for the SFP

Direct Seismic Damage

Initiating events for the SFP were identified in the same way as for reactor faults.

Seismically induced Loss of SFP inventory (sloshing)

The seismic PSA for SFP modelled SFP sloshing as one of the additional failure mechanisms of "Loss of FPC" when the Peak Ground Acceleration (PGA) is larger than a specific value.

Discussions related to the sloshing are included in Section 25.11.2.6 (1), 25.11.2.6 (2) and 25.11.2.9.

Seismically induced Loss of SFP inventory (seismically induced drop effect)

In the internal events SFP PSA, the spent fuel storage rack, Fuel Handling Machine (FHM) and Reactor Building Overhead Crane (RBC) itself were not considered. However, these components have the potential to cause a loss of inventory because of the following reasons. Therefore, these components have been included in the seismic PSA model for SFP.

Spent fuel storage rack: the bottom of the rack is connected (bolted) to the SFP floor. If the bottom of the rack is failed by seismic motion, it also fails the liner sealing of the SFP. Therefore, fuel storage rack attachment failure is considered as a component with the potential to cause a loss of inventory.

FHM and RBC: These components do not fall into the SFP for any internal events. Extremely high magnitude of an earthquake may break the structures for these components and drop it into the SFP. This was modelled in the seismic PSA based on the duration of the cranes remaining just above the SFP.

The SSET for the SFP initiating event is shown in Figure 25.11.2-2.

The headings of many of the SFP SSET are the same as that for the reactor at power seismic SSET. Below is an explanation those unique to the SFP:

Heading "SI-SFP"

This heading denotes the structure of the SFP. Structural failure of the SFP liner likely results in fuel damage. Failure of the liner is assumed to cause an unmitigated sequence leading to fuel damage.

Heading "SLOSHING"

This heading denotes loss of FPC due to sloshing effects. When the PGA is larger than a specific value, FPC is modelled as failed due to sloshing. The failure path of this heading is transferred to the "Loss of SFP cooling" ET (ET name: S-L_COOL). In this case, restart of the FPC pumps is not credited.

The SSET end points are defined as OK (success) or fuel damage sequences, or are transferred to other event trees. The SSET end points are summarised as following.

SE-SFP01	:	Success sequence
S-L_COOL	:	Transferred to "S-L_COOL" ET to address the loss of SFP cooling.
S-LOOP_SFP	:	Transferred to "S-LOOP_SFP" ET to address the LOOP.
S-SBO_SFP	:	Transferred to "S-SBO_SFP" ET to address the LOOP with loss of Class 1AC power.
SE-SFP02	:	Fuel damage sequence.
S-L_INVENTORY	:	Transferred to "S-L_INVENTORY" ET to address the Loss of SFP inventory. The large leak scenario goes into fuel damage sequence.

(c) Initiating event for shutdown

For the discussion of seismic risks for the shutdown conditions, the following considerations were included in a semi-quantitative approach. "Semi-quantitative" means the application of a simpler method such as no consideration of success paths in the quantification.

The following initiating event groups were identified for the shutdown internal events PSA.

- Loss of decay heat removal
- LOOP
- LOCA (including various elevations)

The support system event trees for the shutdown seismic PSA were defined in the same way as in the at power seismic PSA model and no initiating event was assumed to occur unless LOOP occurs. Initiating events that were originally caused by human errors, for example leakage from the Reactor Pressure Vessel (RPV) bottom drain, were excluded since they are not caused by earthquake.

The following are additional seismic effects excluded in the internal events PSA:

(1) Loss of inventory due to structural failures of the operating deck or the SFP

Fragility for the structure or SFP is the same as that of the reactor building. The reactor building fragility has been modelled in the seismic shutdown PSA and the seismic SFP PSA.

(2) Loss of inventory due to failure of the reactor well sealing function

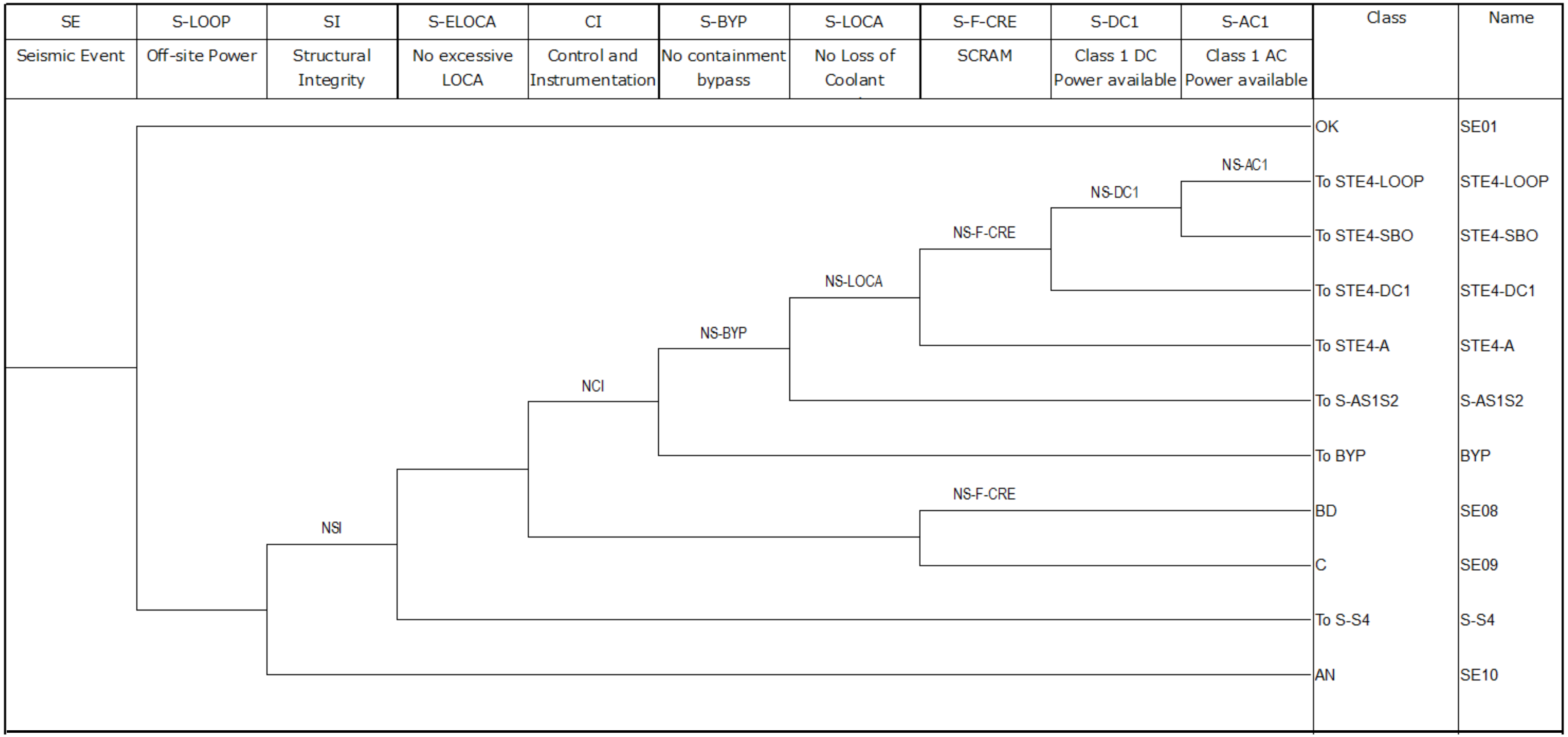


Figure 25.11.2-1 Support System Event Tree (Reactor)

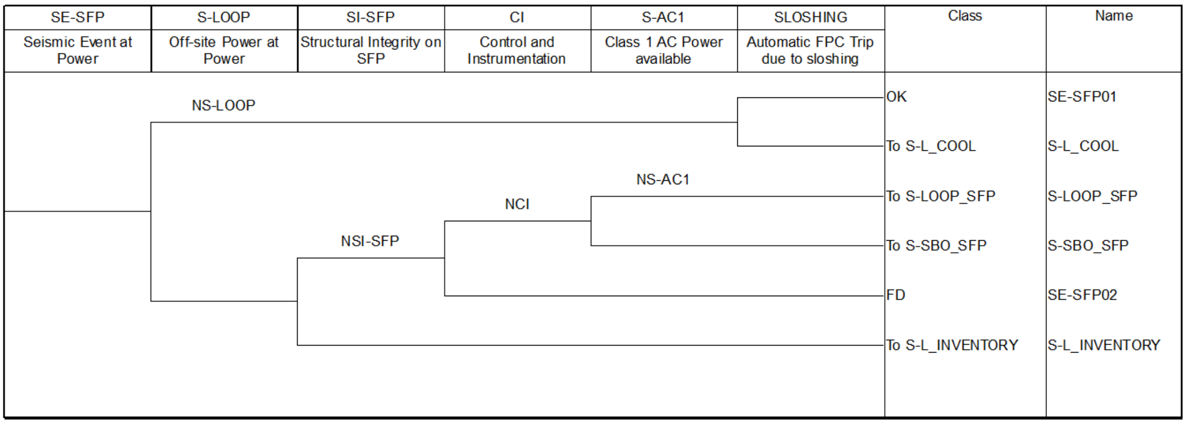


Figure 25.11.2-2 Support System Event Tree (SFP)

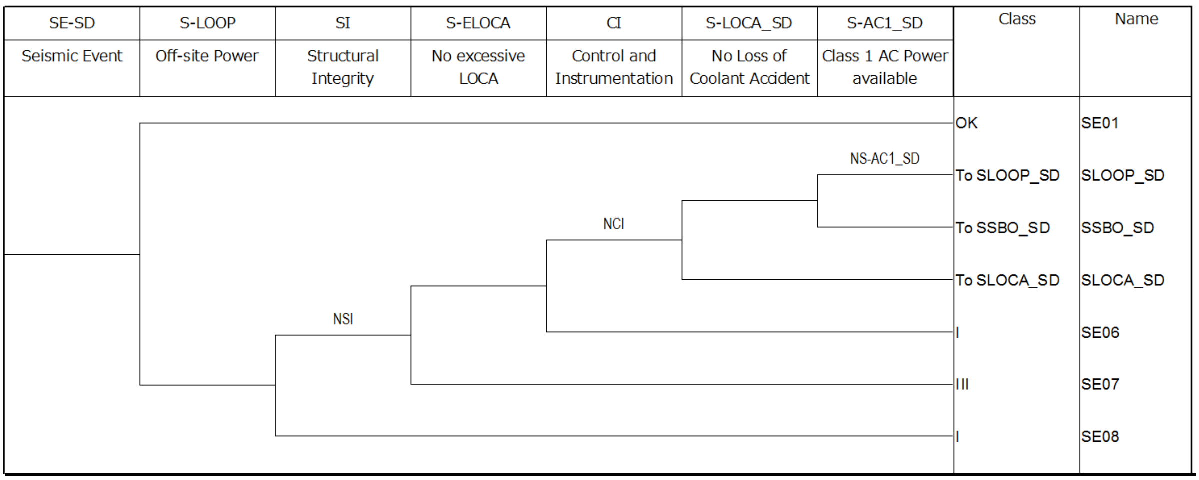


Figure 25.11.2-3 Support System Event Tree (Shutdown)

(2) Event Sequence Analysis

(a) General

The approach for the development of event sequences for the seismic PSA is the same for the reactor at power PSA, shutdown and SFP operations. First, a support system event tree was developed. Then each accident sequence was transferred into the LOOP ET or other ETs. The individual paths through the event trees represent the accident sequences which are then input to the CDF, FDF or LRF analysis.

The event trees except for SSET were developed based on internal events at power PSA models. Basically, the same event trees in the internal events PSA models have been used in the seismic PSA as the event sequence analysis (including success criteria) was the same. Several event trees were slightly changed because of characteristics of this seismic PSA such as the "one fails-all fail" rule or no credit for seismic category 3 SSCs.

An example of an event tree used in the UK ABWR seismic PSA analysis is shown in Figure 25.11.2-4. The individual paths which lead to an end state in an event tree represent the accident sequences which are inputs to the CDF, FDF or LRF analyses.

In the internal events PSA, the success probability can be considered as 1 because the failure probability is sufficiently small. In case of seismic PSA, the success probability could not be assumed as 1 because the failure probability is not sufficiently small for the larger seismic initiating events with high acceleration. Therefore, success sequence fault trees (FTs) have been modelled in the seismic PSA.

(b) Accident sequence definition / event trees for reactor at power

Seismic-induced initiating events were analysed and categorised into a hierarchical event tree (support system event tree).

The following present the explanation of support system event trees and the associated transferred event trees.

As mentioned above, the ETs are basically the same as in the IEAP PSA. All the headings in the support system event trees or changes in the event tree headings from the internal events PSA are explained in this section.

The success criteria are basically the same as the internal events at power PSA for the reactor. However, the required number of redundant systems was not considered for seismically-induced failures because this seismic PSA uses the one fails-all fail rule.

(c) Accident sequence definition / event trees for SFP

The approach for the development of event sequence for the SFP seismic PSA is the same as for the reactor PSA. At first, a support system event tree is developed and then each accident sequence is transferred into the LOOP ET or other ETs.

The event trees except for SSET are developed based on internal events PSA models. Basically, the same event trees as those in the internal events PSA models are used in the seismic PSA because the event sequence analysis (including success criteria) is the same. In the internal events PSA, seismic category 3 systems such as MUWC, SPCU (Suppression Pool Clean-up System) and Fire protection systems are credited. Event trees in the seismic PSA are changed from the internal events PSA because of characteristics of this seismic PSA such as the "one fails-all fail" rule or no credit of seismic category 3 SSCs.

The success probability treatment is the same as in the reactor PSA. No recovery actions are credited in this seismic PSA as well as in the reactor PSA.

(d) Accident sequence definition / event trees for Shutdown

The approach for the development of event sequences for the shutdown seismic PSA is the same as that for the reactor at power. In the internal events PSA, seismic category 3 systems such as MUWC, SPCU and fire protection systems were credited. Event trees in the seismic shutdown PSA have been changed from the internal events PSA because of the characteristics of this seismic PSA, such as the "one fails-all fail" rule and no credit for seismic category 3 SSCs.

The success probability was not modelled in the shutdown seismic PSA because this calculation was performed as a semi-quantitative approach. In this approach, the fragility data prepared for reactor at power or SFP can be commonly used.

No recovery action was credited in this seismic shutdown PSA as in other seismic PSAs.

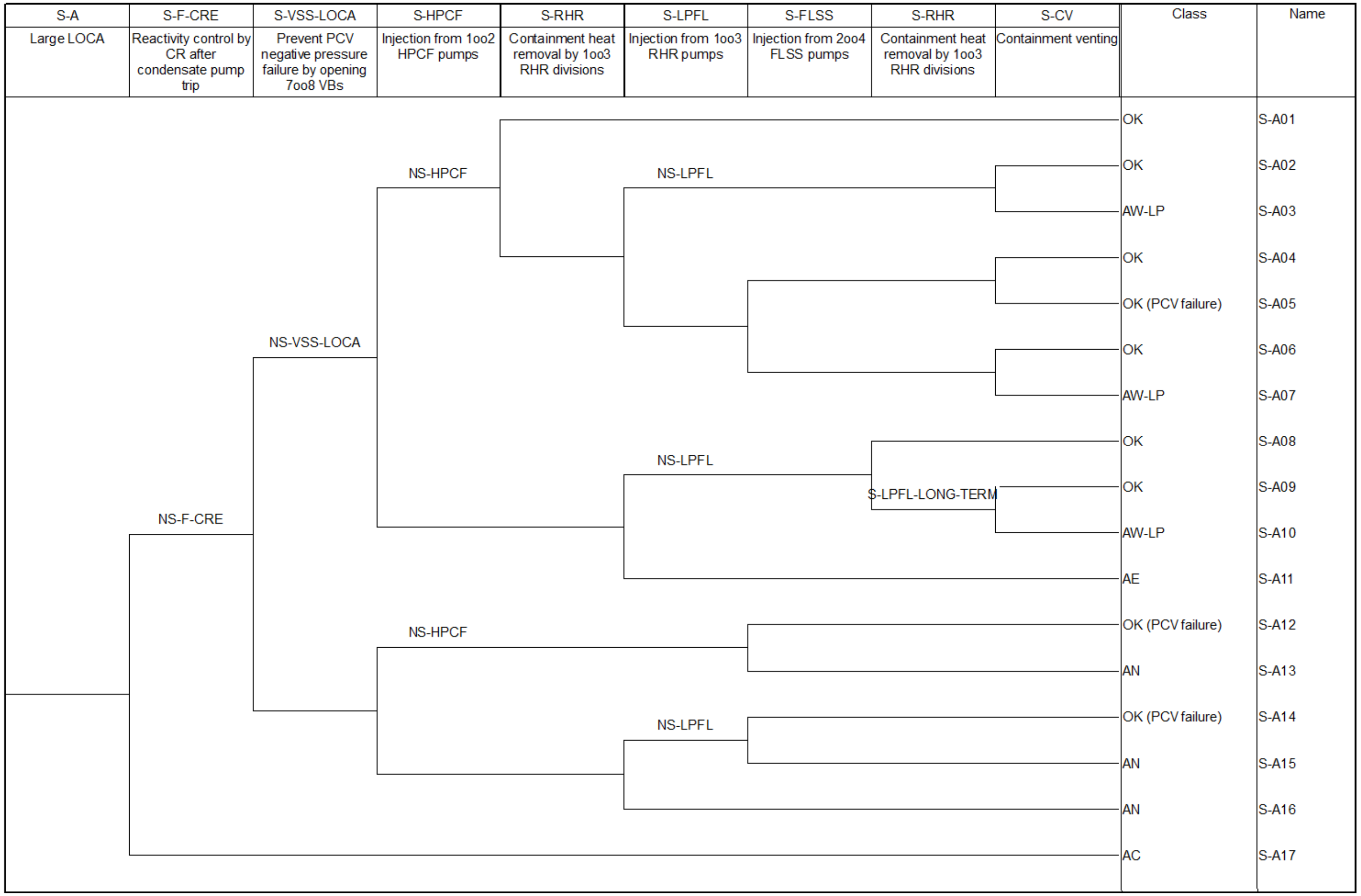


Figure 25.11.2-4 Seismic PSA Event Tree

(3) System Analysis

The seismic system analysis calculated the probability of seismic failure and the corresponding system HCLPFs for each of the important systems throughout the seismic ground acceleration spectrum. The system HCLPFs were then used as inputs to the event trees and combined with random system failure probabilities and human errors. The seismic fault trees contain only those components that might be subject to seismic failures.

Random system failure probabilities were taken from the internal events or SFP analysis and included all other components. One of the important ground rules of the seismic PSA is that the same type components in a system always fail at the same seismic motion.

Fault tree development

Fault trees were developed to support each of the top-level events in the support system event trees discussed above. The fault trees used in the seismic PSA were developed from the same fault trees used in the internal events analysis. The seismic PSA fault trees addressed both random failures and seismically-induced failures.

Seismically-induced failures for each component, building and structure have been newly modelled as fault tree basic events. Examples of component failure modes modelled for fault tree are given as follows. If the component failure can lead to core damage, it is modelled in the fault tree as a basic event.

- Structural failure of pipes (including pipe supports)
- Structural (or Functional) failure of valve (Motor operated valve, Check valve, SRV etc.)
- Structural (or Functional) failure of electric panel or battery rack containing I&C for safety system
- Structural (or Functional) failure of Static structure (Tank, Heat exchanger, Core internal)
- Structural (or Functional) failure of pumps
- Structural failure of Building & Civil structure (R/B)
- All components are in addition considered for random failure

  Random failures in each system were developed from the fault trees of the IEAP Level 1 PSA. Random failures considered in basic events are:

  - Failure or loss of function including out of service due to test or maintenance
  - Human failure
  - CCF (Common Cause Failure)

At the start of the seismic PSA system evaluation, each system was examined to identify any structures or components that had not been evaluated in the internal events PSA but might have a seismic failure mode, which could impact the result. Typically, these are limited to passive components such as piping and buildings which would not be expected to fail in normal operation but might fail as a result of a seismic event.

The evaluation has been simplified by the so-called "one fails-all fail" assumption. This assumption effectively defeats the redundancy built into the design since all identical components in similar locations in a system fail simultaneously as a result of a seismic event. Thus, for example, if a piping in the RHR system is found to fail during a seismic event then all three divisions of RHR are presumed to be affected by the same failure.

The occurrence of an earthquake large enough to cause a LOOP was modelled conservatively in the seismic PSA. Therefore, all fault trees are conditioned on this event.

Based on the above approach, the IEAP Level 1 PSA fault trees, IEAP Level 2 PSA fault trees, SFP Level 1 PSA fault trees and SFP Level 2 PSA fault trees have been reviewed to identify all seismically initiated failures that should be modelled.

For all fault trees in the seismic PSA, seismically-induced failure events were added in and fragility data was applied to each basic event for the quantification. These added basic events have a "Fragility Group ID". This ID enables a one-to-one correspondence between basic event and fragility data for the seismic PSA quantification.

Specific Shutdown considerations

One of the features of the shutdown PSA is that components and systems under maintenance were identified and the outage of the safety systems was considered in each POS. In the seismic PSA, "one fails-all fail" principle was assumed for the mitigation systems. Therefore, although redundant systems/trains in standby may exist in a POS, additional mitigation capability was not credited.

In the shutdown PSA, additional water injection systems (i.e., Make Up Water Condensate (MUWC) and SPCU) have been credited compared to the at power PSA. However, these cannot be credited in the seismic PSA because they have seismic category 3 classification. Note that these systems may still be available, although with a low probability.

(4) Human Reliability Analysis

HRA is normally grouped into three types: type A, type B and type C. The Type A Human Failure Events (HFEs) are the same as those for internal events because they consider errors before the initiating event due to an earthquake. Type B is not considered in the seismic PSA because only initiating events due to the earthquake itself are considered. In this section, type C HRA is discussed.

For the reactor seismic PSA, Type C HRA used the same methodology and the same HFEs as the internal events at power PSA except for the Performance Influence Factors (PIF) which would be impacted by a seismic event. These PIFs have been evaluated for the seismic PSA.

For the SFP seismic PSA, the HRA approach is almost the same as for the internal events SFP PSA.

(a) Performance influence factor for seismic human reliability analysis

Earthquake specific PIFs has been considered based on EPRI's guideline [Ref 25-111] to address the following seismic effects.

- Level of earthquake acceleration

  For the UK ABWR seismic HRA, the earthquake severity was divided into two levels. The operators can cope with various seismically induced accident scenarios. However, the stress level is assumed to be higher when the important SSCs are damaged by relatively large earthquake acceleration.

  PIFs are described in [Ref 25-114].

- Location of Action

  In the IEAP PSA, the locations of each operator action have already been identified. For seismic HRA, the locations of the actions have been divided into two places: MCR (Main Control Room) or outside the MCR. For the operator actions outside of the MCR, including the backup building control room, the PSA used the HRA with PIFs described in Table 25.11.2-3 values for "Out of MCR". For the actions in the MCR, the PIFs for "In the MCR" were used.

- Time After Seismic Event

  Time margins have also been identified in the IEAP PSA. The time margins have been divided into three categories as short (time margin is shorter than 20 min), medium (time margin is longer than 20 min but shorter than 60 min) and long (time margin is longer than 60 min). The PIFs were applied to each category. This categorisation considered three operator action types: action which is needed immediately after the initiating event occurs such as ATWS-related operation (action for criticality control), action for core cooling, and action for long term heat removal. This time frame categorisation is the same as the approach used for the Surry NPP as described in EPRI's guidelines [Ref 25-111].

- Additional dependency

  The dependency between operator actions has been identified in IEAP PSA. When additional dependency was identified in the results of the cutset review, this additional dependency was assessed.

(b) Additional operator action credited in SFP seismic PSA

FPC maintains the SFP cooling in both normal conditions and accident conditions in the UK ABWR. The seismic design for the FPC is considered as seismic category 1 and the Filter Demineraliser (F/D) is designed as seismic category 3. Under the earthquake conditions, "Earthquake Cooling Operating Mode" is used to bypass the F/D.

Operator action is needed in the earthquake condition to prepare the above operation mode for the FPC.

(5) Secondary Hazard and Other Related Issue

(a) Seismically-induced fire

The seismic PSA models the impact of a seismic event on structures and individual equipment. Equipment failures can disable the associated systems. The failure of support systems such as AC or DC power can disable multiple systems. In addition, fires resulting from the seismic events have the potential to disable other systems, which have not been damaged by the earthquake directly. In most seismic PSAs, the impact of fire is judged to be small and is not quantified. No standard PSA techniques are available for evaluating seismically-induced fires.

Although quantitative treatment of seismically-induced fires was not conducted, a qualitative treatment of seismic-induced fire has been carried out at the GDA stage. Potential fire ignition sources were identified and screened, and possible improvements to reduce susceptibility to fire were suggested. A more detailed analysis involving walkdowns of as-installed SSCs will be carried out after the construction of the plant.

The results of the review for seismic induced fire issues have been documented, which were based on the seismic fire ignition sources in NUREG/CR-6850 [Ref 25-96]. Some of these sources have been screened out for current operating plants based on the review by Amico, et al. [Ref 25-112]. At the GDA stage, SSCs designed to Seismic Category 1 can be screened out. If an SSC is identified as a potential seismic fire ignition source, some design improvements have been recommended, which included the anchorage design enhancement and the elimination of potential systems interactions. For SSCs not designed as Seismic Category 1, it was recommended that anchorage be designed to withstand the DBE. There were other concerns associated with the potential for lube oil or fuel oil fires, which should be carefully reviewed during the installation of corresponding SSCs. Similarly, there should be procedures in place for hot work in containment and storage of flammable materials.

(b) Seismically-induced flooding

The impact of seismically-induced flood is somewhat similar to that of seismically-induced fire, in the sense that flood has the potential to damage equipment that has not been damaged by the earthquake directly, but can subsequently impact plant's core damage prevention and mitigation capabilities.

Seismically-induced floods can be caused by internal floods, or by dam failures. Dam failures are discussed further in Section 25.11.2.5(6)(d).

In the same way used for seismically-induced fires, the process involved identifying and screening relevant SSCs whose failures may be induced by seismic events. For the evaluation of flood sources, the focus is on the location of flood sources and the potential flood scenarios.

Based on a survey of experts, the Brookhaven National Laboratory research team [Ref 25-113] concluded that the following SSC categories could be screened out:

- Seismically qualified piping,
- Buried tanks,
- Tanks in vaults,
- Buried piping,
- RCS primary system piping, and
- BWR secondary loop piping or PWR secondary side high energy line piping.

Another reference [Ref 25-174] points out that there are some sources that should always be evaluated. Examples include:

- Non-seismically-qualified Fire Protection system piping, and
- Gravity drain sources with high consequences.

Detailed analysis cannot be carried out at the GDA stage. However, where possible, potential seismic-induced flood issues have been identified and handled through design and procedures at this stage. In the same way used in seismically-induced fire, the SSCs that are already designed as Seismic Category 1 can be screened out. The remaining items will be reviewed and analysed after construction of the plant using detailed walkdowns

(c) Chattering Issues

Electrical contact devices used in electric equipment - such as relays, process switches, and process transmitters - are prone to chattering induced by seismic events, and can lead to the loss of functions for important components.

At the GDA stage, detailed contact chatter evaluation as part of seismic quantification is not possible and should be performed in the post GDA phase.

Electro-mechanical components in safety-class circuits have been qualified and tested for the design basis level earthquake. As long as the plant design precludes seismic interactions (such as cabinet-to-cabinet or cabinet-to-wall bumping), the seismic capacity for these components is given by the "function during" median capacity of the associated cabinets, whether switchgear, MCC, cabinet or panel. All of the safety equipment that is credited in the seismic PSA is seismic Category 1 or 2, and therefore the "function during" seismic capacity of the cabinets can be used for the evaluation of the component chattering issues.

At the GDA stage, it is considered that most of the contact chattering effects can be avoided through good design, as well as ensuring accessibility for operator recovery of chattering locally for feasible operator recovery actions (feasible from the perspective of sufficient time available to perform the action, cues to diagnose the need to perform the recovery, as well as procedural direction). Electrical circuit design will employ solid state devices to the extent possible (e.g., solid state relays, solid state inverters) to avoid chattering effects. Where use of electro-mechanical relays is unavoidable, they will be seismically qualified to the requirements of the particular seismic qualification criteria. Additional design principles to avoid chatter will also be followed, such as bolting together adjacent electrical cabinets and providing stiff anchorage.

(d) Other Secondary effect

(i) Masonry block wall

The seismically induced secondary impacts related to the walls are examined. The walls that have potential impact on the safety case (e.g., initiating event may occur or mitigation systems may be damaged) were evaluated for prevention of wall failure.

(ii) Dam failure

Traditionally, upstream dam failure is considered in the design basis flood evaluation. Downstream dam failure could affect the availability of the ultimate heat sink depending on the design-approach and site-specific features (e.g., coastal site, inland site with cooling pond or cooling tower). These will be addressed quantitatively in the PSA at the post GDA phase.

(6) Level 2 PSA

In the Level 2 seismic PSA, both containment functions and the accident progression sequences have been quantified for the at power condition of the reactor and SFP. The level of detail for this seismic Level 2 PSA is the same as for the internal events PSA except for shutdown conditions. CETs structured in the internal events PSA have been used for the seismic Level 2 PSA, which are identified below.

(a) Interface between L1 and L2

In the IEAP Level 2 PSA for the reactor, a series of CETs were used to quantify all of the accident progression sequences. In the seismic PSA, the same treatment for the interface between the Level 1 and Level 2 PSA was applied. Table 25.11.2-4 shows the end states of the Level 1 seismic PSA event trees (accident class) for the reactor and their associated CETs. The SSET is a special event tree for seismic PSA and the core damage sequence from each SSET end state has been allocated to the appropriate CET.

Accident class CI

This accident class assumes that all functions of C&I are lost by the earthquake and it results in core damage. Since the function to process the initiation signal of the emergency diesel generator (EDG) is also lost after the LOOP, "TBD" is assigned to the PDS for this accident class.

Accident class BYP

This accident class assumes that PCV bypass (in particular, Break Outside Containment (BOC)) occurs due to the earthquake and it results in core damage when the success criteria are not met. According to the same scenario in the internal events PSA, "S3E" or "S3UX" (depends on the event sequence in the transferred event tree) are assigned to the PDS of this accident class.

<u>Accident class E-LOCA</u>

This accident class assumes that excessive LOCA occurs due to the earthquake and it results in core damage. According to the same scenario in the internal events PSA, “S4” or “AN” (depends on the event sequence in the transferred event tree) are assigned to the PDS for this accident class.

<u>Accident class SI</u>

This scenario assumes that a function of water injection is lost due to the earthquake and it results in core damage. Though it is difficult to describe the scenario in detail, “TQUX” (which assumes failure of all front-line systems) is assigned to the PDS for this scenario.

(b) CET development

The CETs used for the Level 2 quantification in this seismic PSA have the same structure as those presented in the IEAP PSA [Ref 25-10].

An example of the CETs for the seismic Level 2 PSA is shown in Figure 25.11.2-5.

(c) Interface between Level 1 and Level 2 PSA for SFP

In the IEAP Level 2 PSA for SFP, four CETs were used to quantify all of the accident progression sequences [Ref 25-76]. In the seismic PSA, the same treatment for the interface between the Level 1 and Level 2 SFP PSA has been applied.

<u>Accident class CI</u>

This accident class assumes that all functions of C&I are lost due to the earthquake and it results in fuel damage in the SFP. The function to process the initiation signal of the EDGs is also lost under the LOOP. This scenario does not include the “loss of inventory” scenario because structural integrity related to loss of inventory is considered in the previous heading. Therefore, “Boil-off” is assigned to the PDS for this accident class.

(d) SFP ET development for Level 2 PSA

The ETs used for quantification in this seismic PSA have the same structure as the IEAP Level 2 SFP [Ref 25 -120].

Table 25.11.2-4 Allocation of Reactor CET (1/2)

ET	Class	No.	IEAP CET (PDS)	Note
SE	BD	**1**	TBD	
	AN	**2**	AN	
S-A	AW-LP	**3**	AW-LP_(LOOPA)	
	AE	**4**	AE_(LOOPA)	
	AN	**2**	AN	
	AC	**5**	AC_(LOOPA)	
S-AFWA	-	-	-	Accident classes are covered by S-A.
S-S1BDL	AW-LP	**3**		
	S12W	**6**		
	AE	**4**		
	S12UX	**7**		
	AN	**2**		
	S12C	**8**		
S-S1HPCF	-	-		Accident classes are covered by S-A or S-S1BDL.
S-S1RHR	-	-		Accident classes are covered by S-A.
S-S2	-	-		Accident classes are covered by S-A.
S-S2RHRB	-	-		Accident classes are covered by S-S1BDL.
S-S2RHRC	-	-		Accident classes are covered by S-A.
S-S2RWLA	-	-		Accident classes are covered by S-A.
S-S2RWLB	-	-		Accident classes are covered by S-A.
S-S2RWLC	-	-		Accident classes are covered by S-A.
S-S2RWLD	-	-		Accident classes are covered by S-A.
S-S2SLC	-	-		Accident classes are covered by S-A.
S-S3BCUW	S3E	**9**		
S-S3BMSFW	-			Accident classes are covered by S-S3BCUW.
S-S3BRCIC	-			Accident classes are covered by S-S3BCUW.
S-S3BSAM	S3E	**9**		
	S3UX	**10**		
S-S4	AN	**2**		
	S4	**11**		
STE4-A	W	**12**	TW_(LOOPA)	
	W-LP	**13**	TW-LP_(LOOPA)	
	C-LP	**14**	TC-LP_(LOOPA)	
	C-HP	**15**	TC-HP_(LOOPA)	
	C	**16**	TC_(LOOPA)	
	CN	**17**	TCN_(LOOPA)	
	AC	**5**	AC_(LOOPA)	

Table 25.11.2-4 Allocation of Reactor CET (2/2)

ET	Class	No.	IEAP CET (PDS)	Note
STE4-DC1	QUV	**18**	TQUV_(C1SBO)	Class1 AC power is lost
	QUX	**19**	TQUX_(C1SBO)	Class1 AC power is lost
	BU	**20**	TBU_(C1SBO)	Class1 AC power is lost
	BD	**1**	TBD	
	NQUV	**21**	TNQUV_(C1SBO)	Class1 AC power is lost
	NQUX	**22**	TNQUX_(C1SBO)	Class1 AC power is lost
	BPN	**23**	TBPN	
	BDN	**24**	TBDN	
	AE	**4**	AE_(C1SBO)	
	AN	**2**	AN	
STE4-HCTL	-	-		Accident classes are covered by STE4-A or STE4-DC1.
STE4-LOOP	W-LP	**13**	TW-LP_(LOOPA)	
	QUV	**18**	TQUV_(LOOPA)	
	QUX	**19**	TQUX_(LOOPA)	
	CN	**17**	TCN (LOOPA)	
	NQUV	**21**	TNQUV_(LOOPA)	
	NQUX	**22**	TNQUX_(LOOPA)	
	AW-LP	**3**	AW-LP_(LOOPA)	
	AE	**4**	AE_(LOOPA)	
	AN	**2**	AN	
STE4-SBO	QUV	**18**	TQUV_(C1SBO)	Class1 AC power is lost
	QUX	**19**	TQUX_(C1SBO)	Class1 AC power is lost
	B	**25**	TB_(C1SBO)	
	BU	**26**	TBU_(C1SBO)	
	BP	**27**	TBP_(C1SBO)	
	NQUV	**21**	TNQUV_(C1SBO)	Class1 AC power is lost
	NQUX	**22**	TNQUX_(C1SBO)	Class1 AC power is lost
	BPN	**28**	TBPN	
	AE	**4**	AE_(C1SBO)	Class1 AC power is lost
	AN	**2**	AN	This class is covered

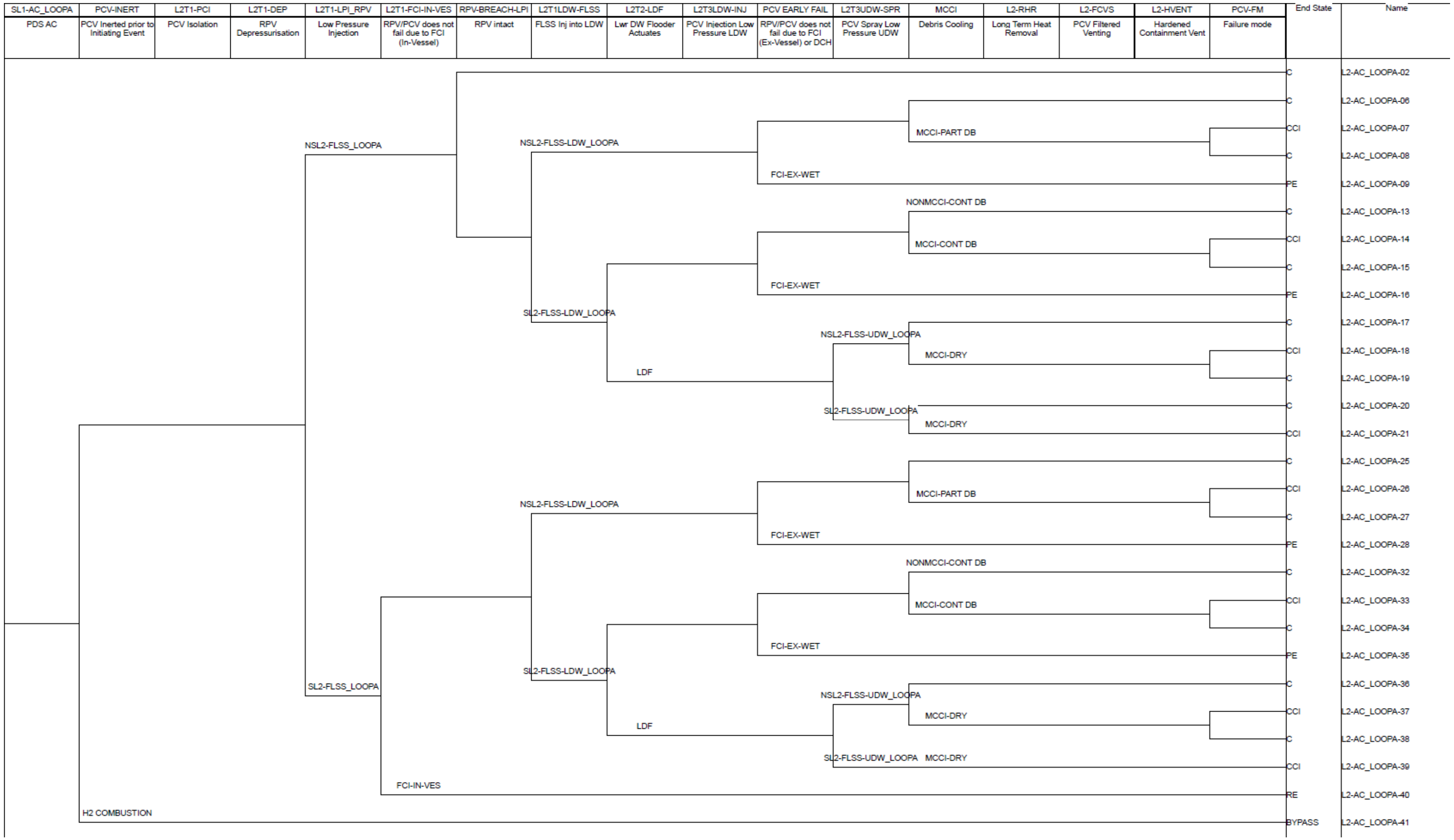


Figure 25.11.2-5 Example of CETs for the Seismic Level 2 PSA

25.11.2.6 Quantification Summary (Level 1 and Level 2)

The UK ABWR seismic PSA model consists of event trees and fault trees that are quantified using a fault tree linking process. The calculation of the total seismic CDF (or FDF/LRF) has been performed as a single top gate.

The total seismic CDF at Power Seismic PSA Level 1 is 7.34E-07 /y and LRF at Level 2 is 6.07E-07 /y.

As for SFP seismic PSA quantification, the total seismic FDF for Level 1 is 4. 51E-07 /y and the seismic LRF for Level 2 is 3.86E-07 /y.

(1) Seismic hazard level result

The results for the at Power Level 1 PSA and Level 2 PSA are shown in Figure 25.11.2-6. The CDF and LRF corresponding to each seismic hazard level are shown. It indicates that the biggest contribution to CDF and LRF comes from seismic ground motions of 0.8 g to 0.9 g. For smaller seismic motions, the LRF is lower than the CDF as expected, since success paths of preventing large release are initiated after core damage. However, for larger seismic events it shows that their values approach each other since systems to prevent core damage and systems to prevent large release fail. Although effort was made to reduce conservatism in the model, the LRF value slightly exceeds the CDF for some higher hazard levels due to the accumulation of conservatism at multiple levels in the complex model.

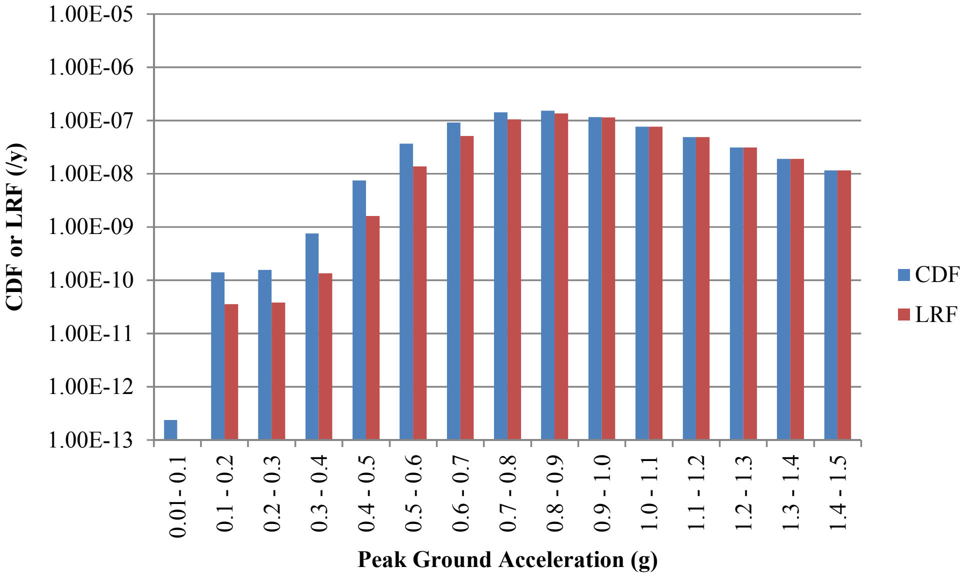


Figure 25.11.2-6 At Power Level 1 and 2 (Seismic CDF and LRF) Results for each Hazard Level

The SFP Level 1 PSA and Level 2 PSA results are summarised in Figure 25.11.2-7. The FDF and LRF corresponding to each seismic hazard level are shown. It indicates that the biggest contribution to FDF comes from 0.8 g to 0.9 g and the biggest contribution to LRF is from 0.8 g to 0.9 g. The sudden increase of the FDF from 0.2 g to 0.3 g to 0.3 g to 0.4 g is due to the sloshing of the SFP water, leading to the loss of FPC.

The Seismic SFP Level 2 PSA models the scenarios with BOC in the reactor side to transfer the sequence to different event trees. The cutsets associated with the reactor side BOC do not have as much of an impact in the Level 1 PSA as in the Level 2 PSA because it fails only FPC, which is not credited over hazard intervals of 0.3 g. Therefore, the modelling complexity expanded the number of cutsets in Level 2 for the larger hazard intervals. If the same number of cutsets had been generated in Level 1 and Level 2 PSAs, the risk contribution from the first group where ACUBE is applied would be significantly smaller for the Level 2 PSA, and thus the CLRP value gets larger than the CCDP in the same hazard interval [Ref 25-115].

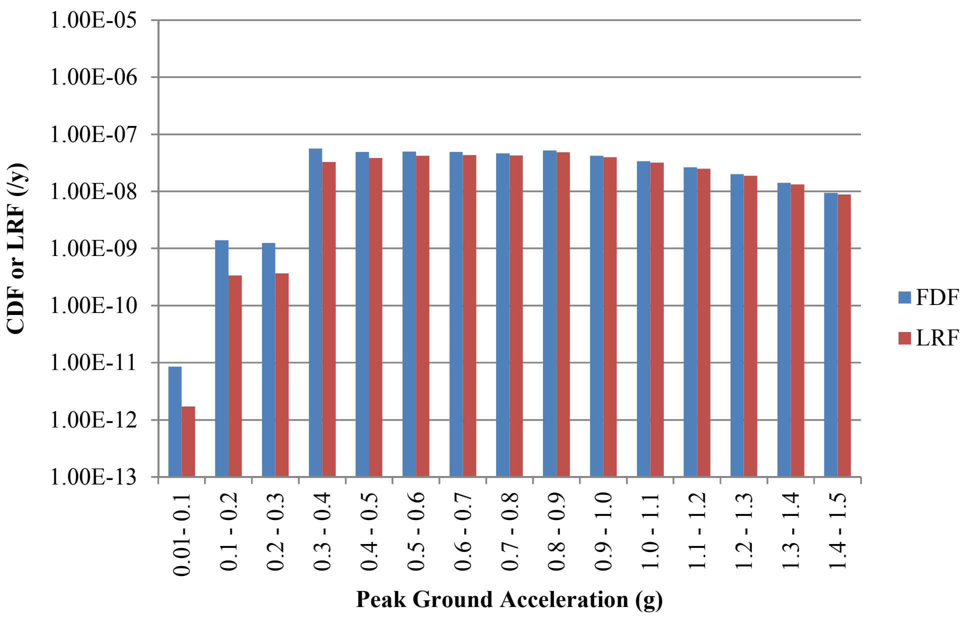


Figure 25.11.2-7 At Power Level 1 and 2 (Seismic FDF and LRF) Results for each Hazard Level

(2) Dominant Sequences

At Power Level 1 PSA for reactor

Figure 25.11.2-8 shows the dominant PDSs for the at Power Level 1 Seismic PSA for the reactor. Figure 25.11.2-9 shows the dominant sequences for the at Power Level 1 seismic PSA for the reactor, which represent the largest risk contribution sequences in terms of all of the plant level fragility. The contribution ratios of PDSs and sequences were obtained by summing the frequencies of each PDS and sequence over each quantified hazard scenario. The frequency was obtained by the product of the CDF obtained by ACUBE and ratio of the sequence or PDS obtained by the CAFTA cutset viewer importance analysis. The events corresponding to each sequence are described in Table 25.11.2-5.

Table 25.11.2-5 The Events Corresponding to each Sequence

Sequence	Events
S-S3BCUW02	This scenario consists of a failure of FLSS after a CUW line break outside containment and its isolation, and success of reactivity control by control rods. It has 24 percent of total CDF.
STE4-SBO04	This scenario consists of a failure of FLSS after a success of AC power supply from Class 2 BBGs and reactor depressurisation by SRVs. It has 18 percent of total CDF.
S-S214	This scenario is a Small LOCA scenario at the Drains and Instrument Taps line. The sequence consists of a failure of FLSS after a failure of ECCS injection and a success of Reactor depressurisation by SRVs. It has 11 percent of total CDF.
STE4-SBO14	This scenario consists of a failure of FLSS after a failure of RCIC injection, and a success of AC power from BBGs as well as reactor depressurisation by SRVs. It has 4.2 percent of total CDF.
S-S1BDL22	This scenario is a Medium LOCA scenario at the Bottom Drain Line. The sequence consists of a failure of FLSS injection after a failure of ECCS injection and a success of reactor depressurisation by SRVs. This sequence has 3.6 percent of total CDF.
STE4-SBO09	This scenario consists of a failure of FLSS after a failure of Class 2 BBGs to provide AC power supply and a success of reactor depressurisation by SRVs. This sequence has 3.4 percent of total CDF.
S-S2RWLD14	This scenario is a Small LOCA scenario at the Reactor water sampling line D. The sequence consists of a failure of FLSS after a failure of ECCS injection and a success of Reactor depressurisation by SRVs. It has a frequency of 1.71E-08 /y and represents 2.3 percent of total CDF.
S-S2RWLC14	This scenario is a Small LOCA scenario at the Reactor water sampling line C. The sequence consists of a failure of FLSS after a failure of ECCS injection and a success of Reactor depressurisation by SRVs. It has 2.2 percent of total CDF.
STE4-SBO5	This scenario consists of a failure of depressurisation by D-ADS after a failure of success of RCIC and AC power by the Backup building generator. It has 2.1 percent of total CDF.
S-S220	This scenario is a Small LOCA scenario at the Drains and Instrument Taps line. The sequence consists of a failure of depressurisation by both ADS and D-ADS after a failure of HPCF injection. It has 1.9 percent of total CDF.

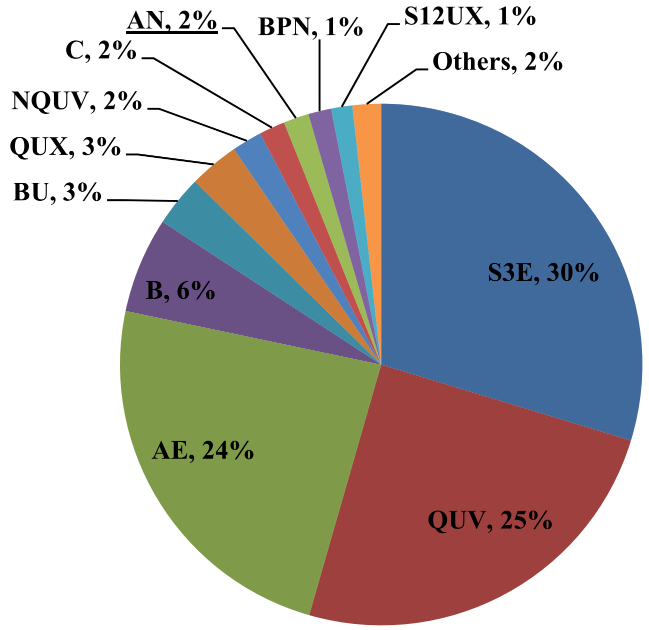


Figure 25.11.2-8 Dominant Plant Damage States (PDS) in Plant Level Fragility for at Power Level 1 Seismic PSA for Reactor

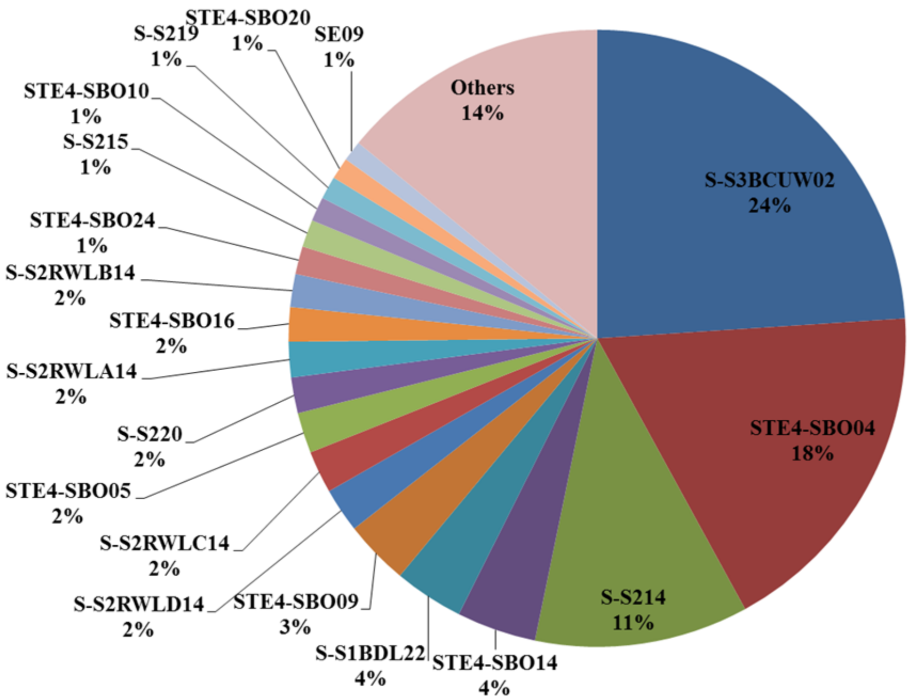


Figure 25.11.2-9 Dominant Sequences in Plant Level Fragility for at Power Level 1 Seismic PSA for Reactor

At Power Level 2 PSA for reactor

Figure 25.11.2-10 shows the dominant sequences for the at Power Level 2 seismic PSA for the reactor with the associated contributions to total seismic LRF. The contribution ratios of sequences were obtained by summing the frequency of a sequence over each quantified hazard scenario. The frequency was obtained by the product of the LRF obtained by ACUBE and the ratio of the sequence obtained by the CAFTA cutset viewer importance analysis.

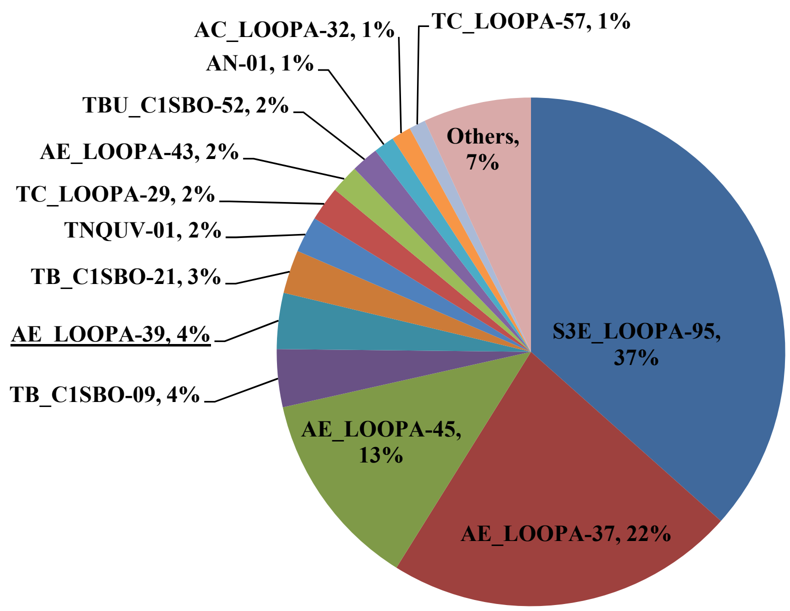


Figure 25.11.2-10 Dominant Sequences in Plant Level Fragility for at Power Level 2 for Reactor

S3E_LOOPA-95: The PDS “S3E” is ISLOCA or BOC with a failure of RPV injection, resulting in low pressure core damage in the short term with containment bypass. This accident sequence corresponds to failure of isolation of the break outside of containment following a LOOP. It contributes 37 percent to the total seismic LRF.

AE_LOOPA-37: This sequence consists of a failure of low pressure injection to the RPV followed by a failure of FLSS injection into the lower drywell, a success of lower drywell flooder and a failure of PCV low pressure spray to the upper drywell and a success of Debris cooling. This sequence contributes 22 percent to the total seismic LRF.

AE_LOOPA-45: The PDS “AE” denotes a LOCA with the failure of RPV injection, resulting in low pressure core damage in the short term. This sequence corresponds to a failure of PCV isolation following a LOOP. This sequence contributes 13 percent to the total seismic LRF.

TB_C1SBO-09: The PDS “TB” is defined by the loss of Class 1 and Class 2 AC power. In this accident sequence, low pressure injection to the RPV fails, but the DAG can be used meaning that there is no failure of FLSS injection and lower drywell flooder actuation succeeds. This accident sequence contributes 3.7 percent to the total seismic LRF.

AE_LOOPA-39: This sequence consists of a failure of low pressure injection to the RPV followed by a failure of FLSS injection into the lower drywell, a success of lower drywell flooder, a success of PCV low pressure spray to upper drywell and a success of Debris cooling. This sequence contributes 3.6 percent to the total seismic LRF.

TB_C1SBO-21: The PDS “TB” is defined by the loss of Class 1 and Class 2 AC power. In this accident sequence, PCV isolation fails. This accident sequence contributes 2.8 percent to the total seismic LRF.

TNQUV-01: The PDS “NQUV” denotes a failure of the SRV tailpipes with a success of reactor depressurisation. In this sequence, H2 combustion does not occur. This accident sequence contributes 2.3 percent to the total seismic LRF.

TC_LOOPA-29: This sequence consists of successful reactor depressurisation followed by failure of FLSS injection to both the RPV and lower drywell, and a success of maintaining the integrity of the RPV and Debris cooling after the success of LDF actuation. This accident sequence contributes 2.2 percent to the total seismic LRF.

AE_LOOPA-43: This sequence consists of a failure of low pressure injection to the RPV followed by a failure of FLSS injection into the lower drywell and actuation of lower drywell flooder, and a success of PCV low pressure spray to the upper drywell. This sequence contributes 1.8 percent to the total seismic LRF.

TBU_C1SBO-52: The PDS “TBU” is “Loss of offsite power transient (T)”, “failure to start all EDGs (B)” and “failure to start RCIC (U)”. This accident sequence corresponds to a failure of PCV isolation following a loss of Class 1 and Class 2 AC power with failure of RCIC. It contributes 1.7 percent to the total seismic LRF.

SFP Level 1 PSA

Figure 25.11.2-11 shows the dominant sequences for the SFP Level 1 PSA with the associated contributions to the total seismic FDF. The contribution ratios of sequences were obtained by summing the frequency of a sequence over each quantified hazard scenario. The frequency was obtained by the product of the FDF obtained by ACUBE and the ratio of the sequence obtained by the CAFTA cutset viewer importance analysis.

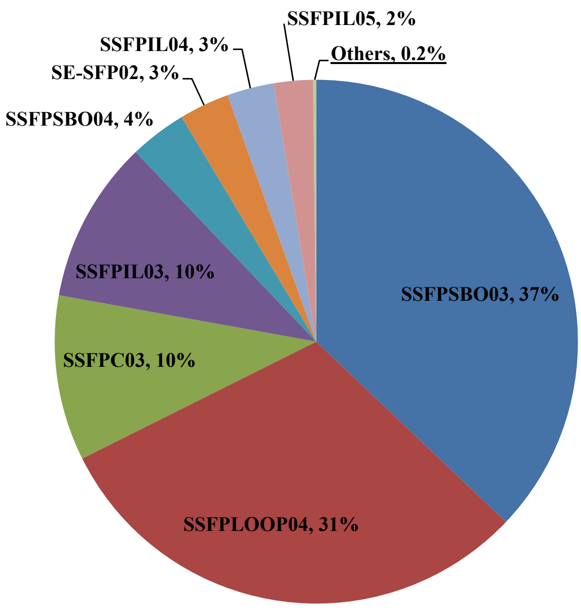


Figure 25.11.2-11 Dominant Accident Sequences for Level 1 SFP

SSFPSBO03: This sequence involves failures of FLSS and FLSR following a successful diagnosis of Loss of SFP cooling. Although the diagnosis has succeeded in detecting the loss of SFP cooling, fuel uncovery eventually occurs resulting in fuel damage. This sequence contributes 37 percent to the total seismic FDF.

SSFPLOOP04: This accident sequence corresponds to the failures of FLSS and FLSR following a successful diagnosis of Loss of SFP cooling after the failure of FPC restart. Although the operator diagnosis has succeeded in detecting the loss of SFP cooling, without make-up, fuel uncovery eventually occurs resulting in fuel damage. This sequence contributes 31 percent to the total seismic FDF.

SSFPC03: This accident sequence corresponds to the failures of FLSS and FLSR following a successful diagnosis of Loss of SFP cooling. Although the operator diagnosis has succeeded in detecting the loss of SFP cooling, without make-up, fuel uncovery eventually occurs resulting in fuel damage. This sequence contributes 10 percent to the total seismic FDF.

SSFPIL03: This accident sequence corresponds to the failures of FLSS and FLSR following successful diagnosis of Loss of SFP inventory. Without make-up, fuel uncovery eventually occurs resulting in fuel damage. It contributes 10 percent to the total seismic FDF.

SSFPSBO04: This accident sequence corresponds to a failure of operator diagnosis on Loss of SFP cooling. Without make-up, fuel uncovery eventually occurs resulting in fuel damage. This sequence contributes 3.5 percent to the total seismic FDF.

SE-SFP02: This accident sequence corresponds to a loss of Control and Instrumentation. This sequence contributes 3.1 percent to the total seismic FDF.

SSFPIL04: This accident sequence corresponds to a failure to diagnose loss of SFP inventory, resulting in a fuel damage end state with 2.9 percent of the total seismic FDF.

SSFPIL05: This accident sequence corresponds to a failure of SFP structural integrity, resulting in a fuel damage end state with 2.4 percent of the total seismic FDF.

SFP Level 2 PSA

Figure 25.11.2-12 shows the dominant sequences for the SFP Level 2 PSA with the associated contributions to the LRF. The contributions of sequences were obtained by summing the frequency of a sequence over each quantified hazard scenario. The frequency was obtained by the product of the LRF obtained by ACUBE and the ratio of the sequence obtained by the CAFTA cutset viewer importance analysis.

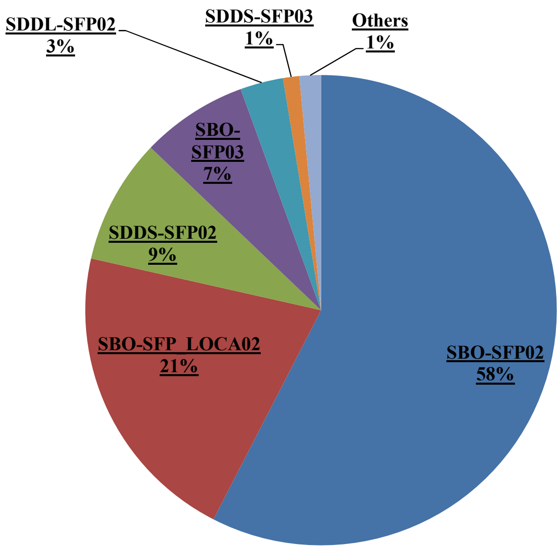


Figure 25.11.2-12 Dominant Accident Sequences for Level 2 SFP

SBO-SFP02: This accident sequence occurs after PDS "Boil-off."

- Fuel uncovery occurs due to boil-off.
- SFP spray after fuel uncovery by FLSS fails.
- Blowout panel is not opened.
- Hydrogen burning does not occur but a significant amount of fission products is released through the leak path of secondary containment with steam.

This sequence contributes 58 percent to the total seismic LRF.

SBO-SFP-LOCA02 is an accident progression sequence after PDS "Boil-off".

- Fuel uncovery occurs due to boil-off. This boil-off scenario is induced by reactor BOC/ISLOCA.
- SFP spray after fuel uncovery by FLSS fails.
- Blowout panel is successfully opened by R/B pressurisation due to BOC/ISLOCA.
- Hydrogen burning does not occur, but a significant amount of fission products is released through the leak path of secondary containment with steam.

This sequence contributes 22 percent to the total seismic LRF.

SBO-SFP03: This sequence is the same as SBO-SFP02 except it ends in hydrogen burning resulting in significant amounts of fission products being released from the damaged location of the secondary containment. This sequence contributes 7.3 percent to the total seismic LRF.

SDDS-SFP02: This accident sequence occurs after PDS "Drain down (small leak)."

- Fuel uncovery occurs due to small leak from SFP.
- SFP spray after fuel uncovery by FLSS fails.
- Blowout panel is successfully opened by R/B pressurisation due to BOC/ISLOCA.
- Hydrogen burning does not occur but a significant amount of fission products is released through the leak path of secondary containment with steam.

This sequence contributes 2.9 percent to the total seismic LRF.

SDDL-SFP02: This accident sequence occurs after PDS "Drain down (large leak)."

- Fuel uncovery occurs due to SFP large leak.
- SFP spray after fuel uncovery by FLSS fails.
- Blowout panel is not opened
- Hydrogen burning does not occur but a significant amount of fission products is released through the leak path of secondary containment with steam.

This sequence contributes 1.1 percent to the total seismic LRF.

(3) Importance Analysis

Importance analysis has been performed with the cutset file containing the results for all seismic hazard levels, with the default CAFTA cutset viewer. The cutset file contains fragility basic events for each group, as presented by "SF-XXXXX-G (hazard interval number)", so that the seismic fragility basic event is considered by unique basic events for the seismic hazard level of concern.

The following subsections discuss top three significant seismic basic events for each Seismic PSA on the dominant hazard interval of 0.8 to 0.9 g.

At Power Level 1 PSA and Level 2 PSA for reactor

Top 3 important seismic basic events are identical in at Power Level 1 PSA and Level 2 PSA for reactor. (The order of CUW Seismic Category 3 Piping and FLSS Tank is changed in Level 2 PSA)

Ceramic Insulator

The seismic basic event with the highest F-V value is "SF-___-CIN". This basic event denotes the failure of the ceramic insulator, a representative Seismic Category 3 component to initiate the Seismic at Power PSA for the reactor.

CUW Seismic Category 3 Piping

The second highest seismic event is "SF-G31-PP", which initiates a BOC at the CUW line with the failure of isolation. CUW system contains two different seismic category areas; the system is categorised in Seismic Category 1 within containment, and Seismic Category 3 outside of the RCCV. The pipes in these different portions of systems are independently modelled, and the failure of the Seismic Category 3 pipe is modelled by this seismic BE. In high seismic hazard intervals, the damage sequence of a BOC at the CUW line represents a large portion of the total frequency such that the importance measure of this component is high.

FLSS Tank

The third highest seismic event is "SF-E71-TKU", which denotes the failure of the FLSS Tank. Most of the accident sequences attempt FLSS injection, such that the F-V importance of this system is high.

SFP Level 1 PSA

FLSS Tank

The highest seismic event is "SF-E71-TKU", which denotes the failure of the FLSS Tank. Most of the accident sequences require FLSS injection, such that the F-V importance measure of this system is high.

Ceramic Insulator

The second highest seismic basic event is "SF-___-CIN". This basic event denotes the failure of the ceramic insulator, a representative Seismic Category 3 component. Since the initiating event starts without a failure of the ceramic insulator in the SFP PSA, the importance of this basic event is lower compared to the at Power PSA for reactor.

FLSS (FLSR) Piping

The first highest seismic event is "SF-E71 (E72)-PP", failure of FLSS and FLSR. A failure of either FLSS or FLSR is modelled in the FT as the failure of both FLSS and FLSR systems, and thus the F-V importance values for these different system pipes are identical. In the SFP PSA, most of the sequences attempt the make-up of the SFP by using both FLSS and FLSR. Therefore, the importance of these systems is high. For the high seismic hazard intervals, the damage sequence of a BOC at the CUW line represents a large portion of the total frequency such that the importance of this component is high.

SFP Level 2 PSA

FLSS Tank

The highest seismic event is "SF-E71-TKU", which denotes the failure of the FLSS Tank. Most of the damage sequences attempt FLSS injection, such that F-V importance of this system is high.

CUW Seismic Category 3 Piping

The second highest seismic event is "SF-G31-PP", which initiates a BOC at the CUW line with the failure of isolation. SFP Level 2 PSA considers the scenarios with the reactor side BOC to transfer into different event trees with the BOP heading.

FLSS (FLSR) Piping

The third highest seismic event is "SF-E71 (E72)-PP", failure of FLSS and FLSR. Failure of either FLSS or FLSR is modelled in the FT as the failure of both FLSS and FLSR systems, and thus the F-V importance values on these different systems' pipes are identical. In the SFP PSA, most of the sequences attempt make-up of the SFP using both FLSS and FLSR. Therefore, the importance is high.

Post-Initiator HFEs for at Power Level 1 PSA for reactor

HFE	F-V	Description
FLSR-UNAVAILABLE	4. 55E-01	FLSR (Mobile Injection Facility) Unavailability
___-HFE-DR-OP	1.31E-02	Failure to open doors to reduce room temperature

The most important Post-Initiator HFE in the Seismic at Power Level 1 PSA is "FLSR-UNAVAILABLE". This is because the long term cooling scenario in the SBO condition is the important sequence in the Seismic at Power PSA for the reactor. In the seismic PSA, a multiplication factor has been applied to the original failure probability if the operator action takes place outside of the control room. Thus, the contribution of FLSR-Unavailability is very important.

The second most important Post-Initiator HFE is the operator action to open doors in the emergency electrical equipment room in the Control Building. Since most of the sequences require to start up the EDGs, the high importance of this human failure event is reasonable.

Post-Initiator HFEs for at Power Level 2 PSA for reactor

HFE	F-V	Description
FLSR-UNAVAILABLE	9.40E-02	FLSR (Mobile Injection Facility) Unavailability
DAG	1.78E-02	Loss of DAG
___-HFE-DR-OP	9. 45E-03	Failure to open doors to reduce room temperature
___-HFE-DP-L2	6. 39E-03	Failure of manual RPV depressurisation during transient for Level 2 PSA
___-HFE-FS-L2	5.40E-03	Failure of depressurisation and FLSS initiation for Level 2 PSA

The most important Post-Initiator HFE in the Seismic at Power Level 2 PSA is "FLSR-UNAVAILABLE". The reason for its high F-V value has been discussed above for the Level 1 PSA.

The second most important Post-Initiator HFE is "DAG". In the Level 2 PSA, heat removal from the PCV is attempted by RHR or LPFL powered by the DAG if the end states in the Level 1 PSA come from the SBO conditions. Since many sequences from the Level 1 PSA challenge the availability of RHR with the DAG, this Post-Initiator HFE has a high F-V value.

The third most important Post-Initiator HFE is the failure to open doors. Since most of the sequences attempt to start up the EDGs in the Level 1 PSA, high importance of this human failure event is reasonable.

Post-Initiator HFEs for SFP Level 1 PSA

HFE	F-V	Description
FLSR-SD	5. 09E-01	FLSR (Mobile Injection Facility) Unavailability
___-HFE-FC-FL	2. 12E-02	Failure of manual initiation of FLSS

The most important Post-Initiator HFE is associated with FLSR unavailability. This is because operator attempts to make up SFP using FLSR in most of the accident scenarios. In the seismic PSA, a multiplication factor has been applied to the original failure probability if the operator action takes place outside of the control room. Thus, the contribution of FLSR-SD is very important.

The second most important Post-Initiator HFE is the operator action for FLSS manual initiation. This operator action takes place in the control room and therefore no multiplication factor is applied to the original failure probability in the model.

Post-Initiator HFEs for SFP Level 2 PSA

HFE	F-V	Description
FLSR-SD	5. 28E-01	FLSR (Mobile Injection Facility) Unavailability
___-HFE-FC-FL2	1.97E-02	Failure of manual initiation of FLSS
___-HFE-FC-FL	1. 05E-02	Failure of manual initiation of FLSS

The most important Post-Initiator HFE is associated with FLSR unavailability. This is because operator attempts to make up SFP using FLSR in most of the accident scenarios. In the seismic PSA, a multiplication factor has been applied to the original failure probability if the operator action takes place outside of the control room. Thus, the contribution of FLSR-SD is very important.

The second most important Post-Initiator HFE is the operator action for FLSS manual initiation in the Level 2 PSA, in case FLSS manual initiation is failed in Level 1. This operator action takes place in the Backup Building and therefore, a multiplication factor has been applied to the original failure probability.

The third most important Post-Initiator HFE is the operator action for FLSS manual initiation in Level 1 PSA. This operator action takes place in the control room and therefore, no multiplication factor has been applied to the original failure probability in the model.

(4) Top Accident Progression Sequences in Significant Release Categories for Level 2 at Power Seismic PSA for the Reactor

BYPASS

The top accident progression sequence in the BYPASS release category is L2-S3E_LOOPA-95. This sequence starts from the S3E PDS in Level 1 PSA for a BOC event. S3E scenario directly leads to containment bypass and thus all sequences ended with S3E are linked to this sequence.

RR

The top accident progression sequence in the RR release category is L2-AE_LOOPA-37. This sequence consists of a success of PCV isolation followed by a failure of FLSS injection into the RPV, lower and upper drywell. In-vessel and ex-vessel fuel cooling interactions do not occur, with a success of LDF actuation and debris cooling.

PCI2

The top accident progression sequence in the PCI2 release category is L2-AE_LOOPA-45. This sequence consists of a failure of PCV isolation after the core damage in the Level 1 PSA with the AE Plant damage state.

LTSBO

The top accident progression sequence in the LTSBO release category is L2-TB_C1SBO-09. The sequence consists of a success of RPV depressurisation and a failure of injection with LPFL. In-vessel and ex-vessel fuel cooling interactions do not occur, with a success of LDF actuation.

C

The top accident progression sequence in the C release category is TC_LOOPA-29. This sequence consists of a success of reactor depressurisation followed by a failure of FLSS injection to RPV, upper and lower drywell. In-vessel and ex-vessel fuel cooling interactions do not occur, with a success of LDF actuation and debris cooling.

SPBYP

The top accident progression sequence in the SPBYP release category is L2-TNQUV-01. The scenario includes the avoidance of H_2 combustion.

OP/OT2

The top accident progression sequence in the OP/OT2 release category is L2-AE_LOOPA-39. This sequence consists of a success of PCV isolation followed by a failure of FLSS injection into the RPV, lower and upper drywell. In-vessel and ex-vessel fuel cooling interactions do not occur, with a success of LDF actuation and failure of debris cooling.

CCI2

The top accident progression sequence in the CCI2 release category is L2-AE_LOOPA-43. This sequence consists of a failure of injection by FLSS to the RPV, upper and lower drywell followed by a failure of LDF actuation. In-vessel and ex-vessel fuel cooling interactions do not occur.

(5) Frequency of Each Release Category and Large Early Release Frequency

Figure 25.11.2-13 shows the ratios of significant release categories and Table 25.11.2-6 shows the frequencies of each release category for the at Power Seismic PSA for reactor for all seismic hazards, which also includes the large early release frequencies. The following release categories are all treated as large early releases:

- Early Containment Failure (ALL_IV_except_BYPASS),
- Late Containment Failure (OP/OT_II),
- Late Containment Failure (ALL_V_except_BYPASS),
- Late Containment Failure (OP/OT_PS_I, II and III),
- In-vessel Fuel-Coolant Interaction (RE_II),
- Ex-vessel Fuel-Coolant Interaction (PE_II),
- Direct Containment Heating (DCH_II),

- PCV Isolation Failure (PCI_II),
- Molten Core Concrete Interaction (CCI_II),
- RPV Rupture (ALL_IX_except BYPASS and RR and RR_LD_AE),
- Containment Bypass (ALL VII and BYPASS_except III),
- S/P Bypass (ALL_VIII_except_BYPASS), and
- Long term SBO (Containment Failure w/o Spray _III).

The frequency of each release category was obtained by summing the frequency of each release category over each quantified hazard scenario. The frequency was obtained by the product of the CDF obtained by ACUBE and the ratio of the release categories obtained by the CAFTA cutset viewer importance analysis. The ratio of release categories in the CAFTA cutset viewer was treated as the ratio of release category in ACUBE post processing. With the increasing seismic hazard level, the sum of F-V importance values of the release category tags becomes more than one due to the consideration of success path by the negations of seismic basic events and due to the factored CCDP in CAFTA. Therefore, the degree of difference is larger at higher seismic hazard levels such as 1.5g, where most of the SSCs are failed by seismic motion. The ratio of F-V importance of each release category over the sum of F-V importance was treated as the repartition of release categories in the total frequency.

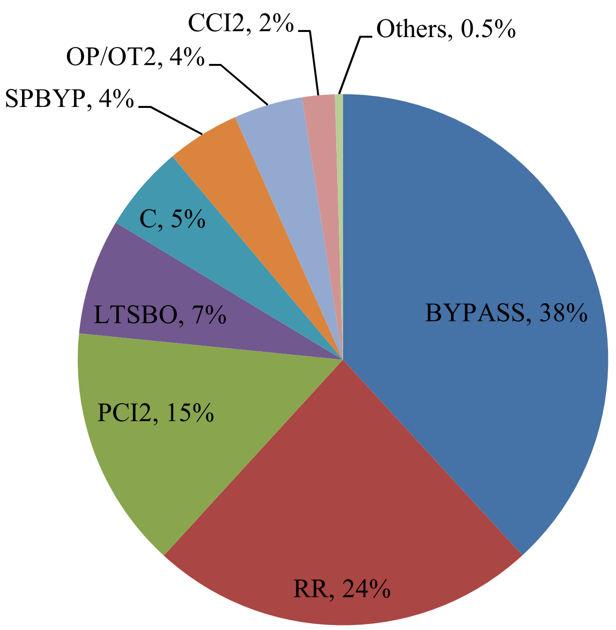


Figure 25.11.2-13 Ratio of Significant Large Release for Level 2 at Power Seismic PSA for Reactor

Table 25.11.2-6 Frequency of each Release Category and Large Early Release Frequency for at Power Seismic PSA for Reactor

No.	Release Category	Code	Frequency (/y)	LRF (/y)	LERF (/y)
1	Containment Leakage (KV and KP_I, II and III)	KP	2. 12E-08	-	-
2	Containment Venting (VV and VP_I, II and III)	VP	2. 53E-09	-	-
3	Filtered Containment Venting (FVV and FVP_I, II and III)	FVP	6.14E-08	-	-
4	Early Containment Failure (ALL_IV_except_BYPASS)	C	3.19E-08	3.19E-08	3.19E-08
5-1	Late Containment Failure (OP/OT_I)	OP/OT1	9.38E-09	-	-
5-2	Late Containment Failure (OP/OT_II)	OP/OT2	2.55E-08	2.55E-08	2.55E-08
5-3	Late Containment Failure (ALL_V_except_BYPASS)	OP/OT3	4.91E-10	4.91E-10	2.44E-10*1
5-4	Late Containment Failure (ALL_VI_except_BYPASS)	OP/OT4	7. 56E-10	-	-
6	Late Containment Failure (OP/OT_PS_I, II and III)	OP/OT_PS	3.19E-09	-	-
7-1	In-vessel Fuel-Coolant Interaction (RE_I)	RE1	6. 49E-12	-	-
7-2	In-vessel Fuel-Coolant Interaction (RE_II)	RE2	5.58E-12	5.58E-12	5.58E-12
8-1	Ex-vessel Fuel-Coolant Interaction (PE_I)	PE1	8.12E-11	-	-
8-2	Ex-vessel Fuel-Coolant Interaction (PE_II)	PE2	2.19E-09	2.19E-09	2.19E-09
9	Direct Containment Heating (DCH_II)	DCH	2.49E-10	2.49E-10	2. 49E-10
10-1	PCV Isolation Failure (PCI_I)	PCI1	1.45E-08	-	-
10-2	PCV Isolation Failure (PCI_II)	PCI2	8.97E-08	8.97E-08	8.97E-08
11-1	Molten Core Concrete Interaction (CCI_I)	CCI1	3.05E-09	-	-
11-2	Molten Core Concrete Interaction (CCI_II)	CCI2	1.20E-08	1.20E-08	1.20E-08
12	RPV Rupture (ALL_IX_except BYPASS and RR and RR_LD_AE)	RR	1.43E-07	1.43E-07	1.43E-07
13	Containment Bypass (ALL VII and BYPASS_except III)	BYPASS	2.32E-07	2.32E-07	2.32E-07
14	S/P Bypass (ALL_VIII_except_BYPASS)	SPBYP	2.70E-08	2.70E-08	2.70E-08
15	Direct Debris Interaction (DDI_II)	DDI	1.11E-08	-	-
16	Long term SBO (Containment Failure w/o Spray _III)	LTSBO	4.28E-08	4.28E-08	3.69E-08*1

Table 25.11.2-7 shows the frequency of each release category for the SFP seismic PSA for all seismic hazards along with the LERF. With the same manner used in at Power Level 2 PSA, the frequency of release category was obtained by summing the frequency of each release category over each quantified hazard scenario. The name of each release category is based on the internal events SFP release categories; an "S" character is added at the beginning of the release category name of the internal events PSA category.

Table 25.11.2-7 Release Categories and Frequencies for Seismic PSA for SFP

No.	Release Category	Frequency (/y)	LRF (/y)	LERF (/y)
1	SDDL-SFP02	1.14E-08	X	X
2	SDDS-SFP02	3.32E-08	X	-
3	SBO-SFP02	2.22E-07	X	-
4	SDDL-SFP03	1.38E-09	X	X
5	SDDS-SFP03	4.32E-09	X	-
6	SBO-SFP03	2.82E-08	X	-
7	SBO-SFP_LOCA02	8.10E-08	X	-
8	SBO-SFP_LOCA03	1.18E-09	X	-
9	SBO-SFP_LOCA04	2.79E-09	X	-
10	SBO-SFP_LOCA05	3.22E-10	X	-
	Total		**3.86E-07**	**1.28E-08**

*1: This Release Category is categorised as LERF in all hazard levels.

25.11.2.7 Assessment of Shutdown PSA

The results for shutdown seismic PSA for the reactor are shown in Figure 25.11.2-14.

Although the quantification included conservatism, the characteristics of the risk contributions for each shutdown Plant Operating States (POS) are similar to those of the internal events PSA. Application of the "one fails-all fail" rule degraded the effect of redundant systems and the risk contribution of POS C (which has an especially large contribution in internal events) was reduced. However, the risk contribution of POS S and POS D (which were insignificant risk contributors in the internal events PSA) became large in the seismic PSA. Redundant systems reduced the risk in internal events PSA; the "one fails-all fail" rule degraded the effect of the redundant systems for the seismic PSA.

The total CDF of seismic shutdown PSA for reactor is about 4.16E-08 /y. This is about 5.67 percent of the CDF under the at power conditions. The duration of a shutdown outage is defined as follows in the internal events shutdown PSA methodology.

> *Initiating event frequencies for the at power PSA were calculated on a reactor calendar year basis. An availability factor of 0.9 was used to convert the reactor critical year frequencies to the reactor calendar year frequencies. To avoid omission or double-counting of internal event risk between the at power PSA and Shutdown PSA, it is assumed that the Shutdown PSA covers 0.1(=1.0-0.9) of a calendar year in average.*

The percentage of shutdown risk contribution is reasonable because of the duration of the shutdown outage described above, the absence of reactivity control accidents, the absence of some seismic specific initiating events in shutdown, and the relatively low decay heat level compared to the at power conditions.

The dominant accident sequence contributing to seismic CDF during shutdown is a seismic-induced loss of offsite power. The Class 1 AC power is not available because RSW is lost. In addition, the injection by FLSS is not available because the FLSS Tank is lost. BOC at CUW occurs as a result of a Seismic Category 3 pipe break with the failure of isolation due to loss of AC power.

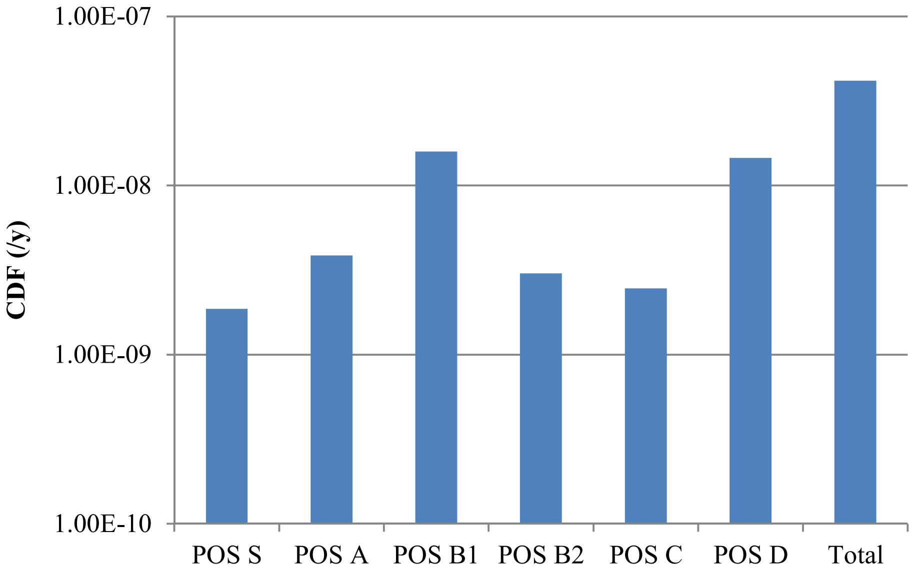


Figure 25.11.2-14 Seismic Shutdown PSA Results for Reactor

25.11.2.8 Level 3 PSA

It is considered that there may be some conservatism in the derivation of frequency for these release categories in the GDA. Therefore, these results should be considered as a conservative assessment of the contribution of external hazards to the combined risk profile. It is expected that this will be reduced in later revisions of the post GDA External Hazards PSAs.

Table 25.11.2-8 and Table 25.11.2-9 show for the reactor at power and the spent fuel pool respectively, the release categories and the corresponding frequencies and representative Level 3 PSA cases.

Table 25.11.2-8 Seismic Event at Power Leading to Fuel Melt Release Categories, Frequencies and Representative Conditional Consequences

Release Categories	Frequency (/y)	Representative Level 3 PSA case
ES1 Containment Leakage from D/W - failed RPV (TQUV)	2.12E-08	P1
ES2 Containment Venting (TQUV no DW sprays)	2.53E-09	P2
ES3 Filtered Containment Venting (TQUV no DW sprays)	6.14E-08	P3
ES4 Early Containment Failure (AC)	3.19E-08	P4
ES5-1 Late Containment Failure (TQUV)	9.38E-09	P5-1
ES5-2 Late Containment Failure (AE)	2.55E-08	P5-2
ES5-3 Late Containment Failure (AW-LP)	4.91E-10	P5-3
ES5-4 late Containment Failure (TW-LP)	7.56E-10	P5-4
ES6 Late Containment Failure with PCV spray (AE)	3.19E-09	P6
ES7-1 In-vessel Fuel-Coolant Interaction (TQUV)	6.49E-12	P7-1
ES7-2 In-vessel Fuel-Coolant Interaction (AE)	5.58E-12	P7-2
ES8-1 Ex-vessel Fuel-Coolant Interaction (TQUV)	8.12E-11	P8-1
ES8-2 Ex-vessel Fuel-Coolant Interaction (AE)	2.19E-09	P8-2
ES9 Direct Containment Heating (TQUX)	2.49E-10	P9
ES10-1 PCV Isolation Failure (TQUV)	1.45E-08	P10-1
ES10-2 PCV Isolation Failure (AE)	8.97E-08	P10-2
ES11-1 Molten Core Concrete Interaction (TQUV)	3.05E-09	P11-1
ES11-2 Molten Core Concrete Interaction (AE)	1.20E-08	P11-2
ES12 RPV rupture (S4)	1.43E-07	P12
ES13 Containment Bypass (S3E)	2.32E-07	P13
ES14 S/P Bypass (TNQUV)	2.70E-08	P14
ES15 Direct Debris Interaction (TQUX)	1.11E-08	P15
ES16 Long Term SBO (TB in-vessel FCI)	4.28E-08	P16
Total Frequency (/y):	**7.34E-07**	

Table 25.11.2-9 Seismic Event Affecting Spent Fuel Pool Release Categories, Frequencies and Representative Conditional Consequences

Release Category	Frequency (/y)	Representative L3 PSA case
SBO-SFP02	2.22E-07	F1+P13
SBO-SFP03	2.82E-08	
SBO-SFP-LOCA02	8.10E-08	
SBO-SFP-LOCA03	1.18E-09	
SBO-SFP-LOCA04	2.79E-09	
SBO-SFP-LOCA05	3.22E-10	
SDDS-SFP02	3.32E-08	F2
SDDS-SFP03	4.32E-09	
SDDL-SFP02	1.14E-08	F3 + P13
SDDL-SFP03	1. 38E-09	
Total frequency /y:	**3. 86E-07**	

(1) Facility Dose Bands

The summated assessment for External Hazards: seismic event leading to fuel melt is given in Table 25.11.2-10. The summated frequency for each facility dose band is presented to enable comparison with the BSO and BSL for Target 8.

- There is no contribution to the 0.0001 to 0.001 Sv (0.1 to 1 mSv) and 0.01 to 0.1 Sv (10 to 100 mSv) facility dose bands.
- There is a negligible contribution to the 0.001 to 0.01 Sv (1 to 10 mSv) facility dose band, at < 0.01 percent of the BSO.
- Release category 3 (identifier ES 3) contributes 6.14E-08 /y to the 0.1 to 1 Sv (100 to 1,000 mSv) facility dose band. This is equivalent to < 1 percent of the BSO.
- The remaining release categories contribute 1.04E-06 /y to the > 1 Sv (>1,000 mSv) facility dose band. This is equivalent to 104 percent of the BSO, or about 1 percent of the BSL and is considered to be in the ALARP region; however, it is noted that there is conservatism in the derivation of External Hazard release category frequencies for the GDA.

Table 25.11.2-10 Assessment Against Facility Dose Bands (Target 8) for External Hazards: Seismic Event Leading to Fuel Melt

Facility Dose Band (Sv)	Release categories assigned to each dose band for IE at power leading to fuel melt	Band Frequency (/y)	Percentage of Band BSO	BSO (/y)	BSL (/y)
0.0001 – 0.001		-	-	1.0E-2	1
0.001 – 0.01	ES1	2. 12E-08	0.0 %	1.0E-3	1.0E-1
0.01 – 0.1		-	-	1.0E-4	1.0E-2
0.1 – 1	ES3	6.14E-08	0.61 %	1.0E-5	1.0E-3
> 1	ES2, ES4, ES5-1, ES5-2, ES5-3, ES5-4, ES6, ES7-1, ES7-2, ES8-1, ES8-2, ES9, ES10-1, ES10-2, ES11-1, ES11-2, ES12, ES13, ES14, ES15, ES16 SBO-SFP02, SBO-SFP03, SBO-SFP-LOCA02, SBO-SFP-LOCA03, SBO-SFP-LOCA04, SBO-SFP-LOCA05, SDDS-SFP02, SDDS-SFP03, SDDL-SFP02, SDDL-SFP03	1.04E-06	104 %	1.0E-6	1.0E-4
Summated frequency of release categories /y		**1.12E-06**			

(a) Contribution from seismic event while reactor at power leading to fuel melt

Table 25.11.2-11 presents the assessment against Target 8 for the reactor at power alone, based on the release category frequencies of Table 25.11.2.8-8 and the facility dose band allocations presented in Section 25.7.

The seismic event affecting the reactor at power alone contributes 6.51E-07 /y to the > 1 Sv (> 1,000 mSv) facility dose band. This is equivalent to 65.1 percent of the BSO.

- Two release categories contribute 3.75E-07 /y or 51.1 percent of the band frequency for the reactor at power and, individually, are > 10 percent of the BSO for this dose band:
  - Release category ES 12, **RPV rupture (S4)**, at 1.43E-07 /y (19.5 percent of the band frequency),
  - Release category ES 13, **Containment Bypass (S3E)**, at 2.32E-07 /y (31.6 percent of the band frequency).
- A further eight release categories contribute 2.55E-07 /y or 34.7 percent of the band frequency for the reactor at power and, individually, are > 1 percent of the BSO for this dose band:
  - Release category ES 10-2, **PCV Isolation Failure (AE)**, at 8.97E-08 /y (12.2 percent of the band frequency),
  - Release category ES 4, **Early Containment Failure (AC)**, at 3.19E-08 /y (4.4 percent of the band frequency),
  - Release category ES 14, **S/P Bypass (TNQUV)**, at 2.70E-08 /y (3.7 percent of the band frequency),

- Release category ES 16, **Long Term SBO (TB in-vessel FCI)**, at 4.28E-08 /y (5.8 percent of the band frequency),
- Release category ES 5-2, **Late Containment Failure (AE)**, at 2.55E-08 /y (3.5 percent of the band frequency),
- Release category ES 10-1, **PCV Isolation Failure (TQUV)**, at 1.45E-08 /y (2.0 percent of the band frequency),
- Release category ES 11-2, **Molten Core Concrete Interaction (AE),** at 1.20E-08 /y (1.6 percent of the band frequency),
- Release category ES 15, **Direct Debris Interaction (TQUX)**, at 1.11E-08 /y (1.5 percent of the band frequency).

Table 25.11.2-11 Assessment Against Facility Dose Bands (Target 8) for External Hazards: Seismic Event Affecting Reactor at power Leading to Fuel Melt

Facility Dose Band (Sv)	Release categories assigned to each dose band for IE at power leading to fuel melt	Band Frequency (/y)	Percentage of Band BSO	BSO (/y)	BSL (/y)
0.0001 – 0.001		-	-	1.0E-2	1
0.001 – 0.01	ES1	2.12E-08	0.0 %	1.0E-3	1.0E-1
0.01 – 0.1		-	-	1.0E-4	1.0E-2
0.1 – 1	ES3	6.14E-08	0.61 %	1.0E-5	1.0E-3
> 1	ES2, ES4, ES5-1, ES5-2, ES5-3, ES5-4, ES6, ES7-1, ES7-2, ES8-1, ES8-2, ES9, ES10-1, ES10-2, ES11-1, ES11-2, ES12, ES13, ES14, ES15, ES16	6.51E-07	65.1 %	1.0E-6	1.0E-4
Summated frequency of release categories /y		**7.34E-07**			

(b) Contribution from seismic event leading to fuel melt in the SFP

Table 25.11.2-12 presents the assessment against Target 8 for the SFP initiators, based on the release category frequencies of Table 25.11.2-9 and the facility dose band allocations presented in Section 25.7. All release categories are allocated to the highest facility dose band.

- The seismic event affecting the SFP contributes 3.86E-07 /y to the > 1 Sv (> 1,000 mSv) facility dose band. This is equivalent to 38.6 percent of the BSO.
- Release category SBO-SFP02 is dominant and contributes 2.22E-07 /y (57.5 percent of the band frequency).
- A further three release categories contribute an additional 1.42E-07 /y or 36.9 percent of the total and, individually, are > 1 percent of the BSO for this dose band:
  - Release category SBO-SFP_LOCA02 at 8.10E-08 /y (21.0 percent of the band frequency),
  - Release category SBO-SFP03 at 2.82E-08 /y (7.3 percent of the band frequency),
  - Release category SDDS-SFP02 at 3.32E-08 /y (8.6 percent of the band frequency).

Table 25.11.2-12 Assessment Against Facility Dose Bands (Target 8) for External Hazards: Seismic Event Seismic Event Affecting the SFP Leading to Fuel melt

Facility Dose Band (Sv)	Release categories assigned to each dose band for IE at power leading to fuel melt	Band Frequency (/y)	Percentage of Band BSO	BSO (/y)	BSL (/y)
0.0001 – 0.001	-	-	-	1.0E-2	1
0.001 – 0.01	-	-	-	1.0E-3	1.0E-1
0.01 – 0.1	-	-	-	1.0E-4	1.0E-2
0.1 – 1	-	-	-	1.0E-5	1.0E-3
> 1	SBO-SFP02, SBO-SFP03, SBO-SFP-LOCA02, SBO-SFP-LOCA03, SBO-SFP-LOCA04, SBO-SFP-LOCA05, SDDS-SFP02, SDDS-SFP03, SDDL-SFP02, SDDL-SFP03	3.86E-07	38.6 %	1.0E-6	1.0E-4
Summated frequency of release categories /y		**3.86E-07**			

(2) Individual Risk

The individual risk for each release category for External Hazards: seismic event affecting reactor at power leading to fuel melt is calculated as the product of the conditional individual risk for the equivalent IE at power release category given in [Ref 25-1] and the release category frequency given in Table 25.11.2-8. The individual risk for each release category for External Hazards: seismic event affecting SFP leading to fuel melt is calculated as the product of the conditional individual risk for the equivalent SFP IE given in [Ref 25-1] and the release category frequency given in Table 25.11.2-9.

The summated individual risk at 1 km from External Hazard: seismic event leading to fuel melt is 1.16E-07 /y or 11.6 percent of the BSO, as given in Table 25.11.2-13.

Table 25.11.2-13 Individual Risk of Fatality Close to the Site (Target 7) for External Hazards: Seismic Event Leading to Fuel Melt

Fault Group	Individual risk of fatal health effects (/y)		
	400 m	1,000 m	1,500 m
External Hazards: seismic event affecting reactor at power	1.06E-07	6.91E-08	5.50E-08
External Hazards: seismic event affecting SFP	6.32E-08	4.70E-08	4.12E-08
Total individual risk (/y):	**1.69E-07**	**1.16E-07**	**9.63E-08**
Total as % of BSO	**16.9 %**	**11.6 %**	**9.6 %**
BSO	**1.00E-06**		
BSL	**1.00E-04**		

(a) **Contribution from seismic event affecting reactor at power leading to fuel melt**

The seismic event affecting the reactor alone contributes 6.91E-08 /y to the individual risk at 1 km. This is equivalent to 6.9 percent of the BSO.

The contribution of each release category to the overall risk from IE at power is given in Table 25.11.2-14 and represented graphically in Figure 25.11.2-15.

- Three release categories contribute 5.38E-08 /y or 77.7 percent of the total from the reactor at power and, individually, are > 1 percent of the BSO at 1km:
  - Release category ES 13, **Containment Bypass (S3E)**, at 2.86E-08 /y (41.3 percent of the total),
  - Release category ES 12, **RPV rupture (S4)**, at 1.51E-08 /y (21.8 percent of the total),
  - Release category ES 10-2, **PCV Isolation Failure (AE)**, at 1.01E-08 /y (14.6 percent of the total).

Table 25.11.2-14 Individual Risk of Fatality Close to the Site (Target 7) for External Hazards: Seismic Event Affecting Reactor at power Leading to Fuel Melt

Release Category	Release Category Frequency (/y)	Individual risk of fatal health effects (/y)			Contribution to Total Risk at 1 km
		400 m	1,000 m	1,500 m	
ES 1	2.12E-08	1.24E-13	4.85E-14	3.01E-14	0.00 %
ES 2	2.53E-09	3.57E-11	1.03E-11	5.74E-12	0.01 %
ES 3	6.14E-08	4.54E-11	1.61E-11	1.05E-11	0.02 %
ES 4	3.19E-08	5.58E-09	3.55E-09	2.84E-09	5.13 %
ES 5-1	9.38E-09	6.89E-10	2.85E-10	1.88E-10	0.41 %
ES 5-2	2.55E-08	3.83E-09	1.99E-09	1.37E-09	2.88 %
ES 5-3	4.91E-10	1.08E-10	7.11E-11	5.65E-11	0.10 %
ES 5-4	7.56E-10	8.16E-11	3.33E-11	2.15E-11	0.05 %
ES 6	3.19E-09	2.47E-10	8.91E-11	5.71E-11	0.13 %
ES 7-1	6.49E-12	1.10E-12	6.04E-13	4.29E-13	0.00 %
ES 7-2	5.58E-12	1.07E-12	6.77E-13	5.18E-13	0.00 %
ES 8-1	8.12E-11	3.94E-12	1.28E-12	7.78E-13	0.00 %
ES 8-2	2.19E-09	3.37E-10	1.84E-10	1.29E-10	0.27 %
ES 9	2.49E-10	3.11E-11	1.36E-11	9.24E-12	0.02 %
ES 10-1	1.45E-08	9.56E-10	3.34E-10	2.09E-10	0.48 %
ES 10-2	8.97E-08	1.55E-08	1.01E-08	7.92E-09	14.6 %
ES 11-1	3.05E-09	2.48E-10	9.12E-11	5.67E-11	0.13 %
ES 11-2	1.20E-08	2.11E-09	1.08E-09	7.43E-10	1.56 %
ES 12	1.43E-07	2.52E-08	1.51E-08	1.11E-08	21.8 %
ES 13	2.32E-07	3.87E-08	2.86E-08	2.46E-08	41.3 %
ES 14	2.70E-08	4.19E-09	2.54E-09	1.87E-09	3.67 %
ES 15	1.11E-08	1.08E-09	4.45E-10	2.87E-10	0.64 %
ES 16	4.28E-08	7.23E-09	4.64E-09	3.58E-09	6.71 %
Total individual risk (/y):		**1.06E-07**	**6.91E-08**	**5.50E-08**	
Total as % of BSO		**10.6 %**	**6.9 %**	**5.5 %**	
BSO		**1.00E-06**			
BSL		**1.00E-04**			

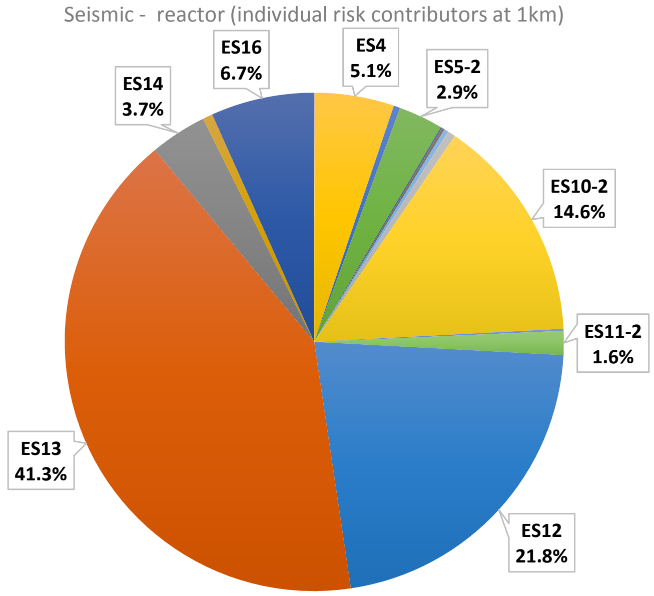


Figure 25.11.2-15 Contribution of Release Categories to the Individual Risk at 1 km for External Hazards: Seismic Event Affecting Reactor at power Leading to Fuel Melt

(b) Contribution from seismic event affecting the SFP leading to fuel melt

The Internal Flood initiators alone contribute 4.70E-08 /y to the individual risk at 1 km. This is equivalent to 4.7 percent of the BSO.

The contribution of each release category to the overall risk from IE at power is given in Table 25.11.2-15 and represented graphically in Figure 25.11.2-16.

- Two release categories contribute 3.73E-08 /y or 79.2 percent of the total from the SFP and, individually, are > 1 percent of the BSO at 1 km:
  - Release category SBO-SFP02, at 2.73E-08 /y (58.0 percent of the total),
  - Release category SBO-SFP_LOCA02, at 9.95E-09 /y (21.2 percent of the total).

Table 25.11.2-15 Individual Risk of Fatal Health Effects near to the Site (Target 7) for External Hazards: Seismic Event Affecting the SFP Leading to Fuel Melt

Release Category	Individual risk of fatal health effects (/y)			Contribution to Total Risk at 1 km
	400 m	1,000 m	1,500 km	
SBO-SFP02	3.66E-08	2.73E-08	2.40E-08	58.0 %
SBO-SFP03	4.65E-09	3.46E-09	3.05E-09	7.37 %
SBO-SFP-LOCA02	1.34E-08	9.95E-09	8.75E-09	21.2 %
SBO-SFP-LOCA03	1.95E-10	1.45E-10	1.27E-10	0.31 %
SBO-SFP-LOCA04	4.60E-10	3.43E-10	3.01E-10	0.73 %
SBO-SFP-LOCA05	5.31E-11	3.95E-11	3.48E-11	0.08 %
SDDS-SFP02	5.28E-09	3.93E-09	3.42E-09	8.36 %
SDDS-SFP03	6.87E-10	5.11E-10	4.45E-10	8.36 %
SDDL-SFP02	1.69E-09	1.20E-10	1.02E-09	2.55 %
SDDL-SFP03	2.04E-10	1.45E-10	1.23E-10	0.31 %
Total individual risk: /y	**6.32E-08**	**4.70E-08**	**4.12E-08**	
Total as % of BSO	**6.3 %**	**4.7 %**	**4.1 %**	
BSO	1.00E-06			
BSL	1.00E-04			

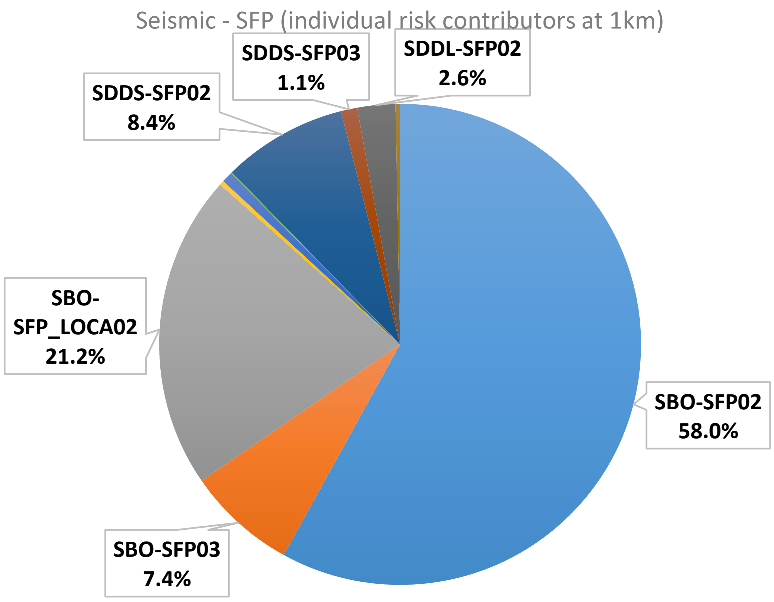


Figure 25.11.2-16 Contribution of Release Categories to the Individual Risk at 1 km for External Hazards: Seismic Event Affecting the SFP Leading to Fuel Melt

(3) Societal Risk

Table 25.11.2-16 presents the assessment against Target 9 for the External Hazards: seismic event leading to fuel melt. This is based on the summated numbers of short term fatal health effects and notional late fatalities in the UK population presented in [Ref 25-1].

All release categories except ES 1 and ES 3 are assigned as above the Target 9 threshold. As all release categories assigned to the top facility dose band currently lead to an expectation of > 100 deaths, with minimal protective actions, the frequency breakdown is the same as that given in sub-section 25.11.2.8.1.

The summated societal risk from External Hazards: seismic event is 1.04E-06 /y, i.e. about 1,040 percent of the BSO or about a factor of 10 above the BSO. This is equivalent to 10.4 percent of the BSL. For the group, the summated frequency is considered to be in the tolerable if ALARP region; however, it is noted that there is conservatism in the derivation of External Hazard release category frequencies for the GDA.

It can be seen from Table 25.11.2-17 and Table 25.11.2-18 that three release categories are individually above the BSO.

Table 25.11.2-16 Frequency of Exceeding the Societal Threshold (Target 9) for External Hazards: Seismic Event Leading to Fuel Melt

Sum of all release categories	Frequency above Target 9 threshold (/y)
External Hazards: seismic event affecting reactor at power	**6.51E-07**
Internal Hazards: seismic event affecting SFP	**3.86E-07**
Total frequency (/y):	**1.04E-06**
Total as % of BSO	**1,040 %**
BSO	**1.00E-07**
Total as % of BSL:	**10.4 %**
BSL	**1.00E-05**

Table 25.11.2-17 Frequency of Exceeding the Societal Threshold (Target 9) for External Hazards: Seismic Event Leading to Fuel Melt with Minimal Offsite Protective Actions

Release Category	Release category frequency (/y)	Frequency above Target 9 threshold (/y)	Contribution to Total Frequency
ES 1	2.12E-08	0.00E+00	0.00 %
ES 2	2.53E-09	2.53E-09	0.39 %
ES 3	6.14E-08	0.00E+00	0.00 %
ES 4	3.19E-08	3.19E-08	4.90 %
ES 5-1	9.38E-09	9.38E-09	1.44 %
ES 5-2	2.55E-08	2.55E-08	3.91 %
ES 5-3	4.91E-10	4.91E-10	0.08 %
ES 5-4	7.56E-10	7.56E-10	0.12 %
ES 6	3.19E-09	3.19E-09	0.49 %
ES 7-1	6.49E-12	6.49E-12	0.00 %
ES 7-2	5.58E-12	5.58E-12	0.00 %
ES 8-1	8.12E-11	8.12E-11	0.01 %
ES 8-2	2.19E-09	2.19E-09	0.34 %
ES 9	2.49E-10	2.49E-10	0.04 %
ES 10-1	1.45E-08	1.45E-08	2.23 %
ES 10-2	8.97E-08	8.97E-08	13.8 %
ES 11-1	3.05E-09	3.05E-09	0.47 %
ES 11-2	1.20E-08	1.20E-08	1.84 %
ES 12	1.43E-07	1.43E-07	22.0 %
ES 13	2.32E-07	2.32E-07	35.6 %
ES 14	2.70E-08	2.70E-08	4.14 %
ES 15	1.11E-08	1.11E-08	1.70 %
ES 16	4.28E-08	4.28E-08	6.57 %
	Total Frequency: /y	**6.51E-07**	
	Total as % of BSO	**651 %**	
	BSO	**1.00E-07**	
	Total as % of BSL	**6.5 %**	
	BSL	**1.00E-05**	

Table 25.11.2-18 Frequency of Exceeding the Societal Threshold (Target 9) for External Hazards: Seismic Event Affecting the SFP Leading to Fuel Melt with Minimal Offsite Protective Actions.

Release Category	Release category frequency (/y)	Frequency above Target 9 threshold (/y)	Contribution to total frequency
SBO-SFP02	2.22E-07	2.22E-07	57.5 %
SBO-SFP03	2.82E-08	2.82E-08	7.31 %
SBO-SFP-LOCA02	8.10E-08	8.10E-08	21.0 %
SBO-SFP-LOCA03	1.18E-09	1.18E-09	0.31 %
SBO-SFP-LOCA04	2.79E-09	2.79E-09	0.72 %
SBO-SFP-LOCA05	3.22E-10	3.22E-10	0.08 %
SDDS-SFP02	3.32E-08	3.32E-08	8.61 %
SDDS-SFP03	4.32E-09	4.32E-09	1.12 %
SDDL-SFP02	1.14E-08	1.14E-08	2.95 %
SDDL-SFP03	1.38E-09	1.38E-09	0.36 %
	Total Frequency: /y	**3.86E-07**	
	Total as % of BSO	**386 %**	
	BSO	1.00E-07	
	Total as % of BSL	**3.9 %**	
	BSL	1.00E-05	

(a) Contribution from seismic event affecting reactor at power leading to fuel melt

Table 25.11.2-17 presents the assessment against Target 9 for the External Hazards: seismic event affecting the reactor at power, based on the release category frequencies of Table 25.11.2-7 and the predicted societal consequences presented in Section 25.7. The release categories contributing to this frequency are shown in Figure 25.11.2-17.

The seismic event affecting the reactor alone contributes 6.51E-07 /y to the frequency of exceeding the Target 9 threshold with minimal protective actions. This is equivalent to 651 percent of the BSO or a factor of 6.5 above the BSO. This is equivalent to 6.5 percent of the BSL.

- Two release categories contribute 3.75E-07 /y or 57.6 percent of the total frequency:
  - Release category ES 13, **Containment Bypass (S3E)**, at 2.32E-07 /y (35.6 percent of the total frequency).
  - Release category ES 12, **RPV rupture (S4)**, at 1.43E-07 /y (22.0 percent of the total frequency),
- A further eight release categories contribute 2.55E-07 /y or 39.0 percent of the total frequency and, individually, are > 10 percent of the BSO:

- Release category ES 10-2, **PCV Isolation Failure (AE)**, at 8.97E-08 /y (13.8 percent of the total frequency),
- Release category ES 4, **Early Containment Failure (AC)**, at 3.19E-08 /y (4.9 percent of the total frequency),
- Release category ES 14, **S/P Bypass (TNQUV)**, at 2.70E-08 /y (4.1 percent of the total frequency),
- Release category ES 16, **Long Term SBO (TB in-vessel FCI)**, at 4.28E-08 /y (6.6 percent of the total frequency),
- Release category ES 5-2, **Late Containment Failure (AE)**, at 2.55E-08 /y (3.9 percent of the total frequency),
- Release category ES 10-1, **PCV Isolation Failure (TQUV)**, at 1.45E-08 /y (2.2 percent of the total frequency),
- Release category ES 11-2, **Molten Core Concrete Interaction (AE),** at 1.20E-08 /y (1.8 percent of the total frequency),
- Release category ES 15, **Direct Debris Interaction (TQUX)**, at 1.11E-08 /y (1.7 percent of the total frequency).

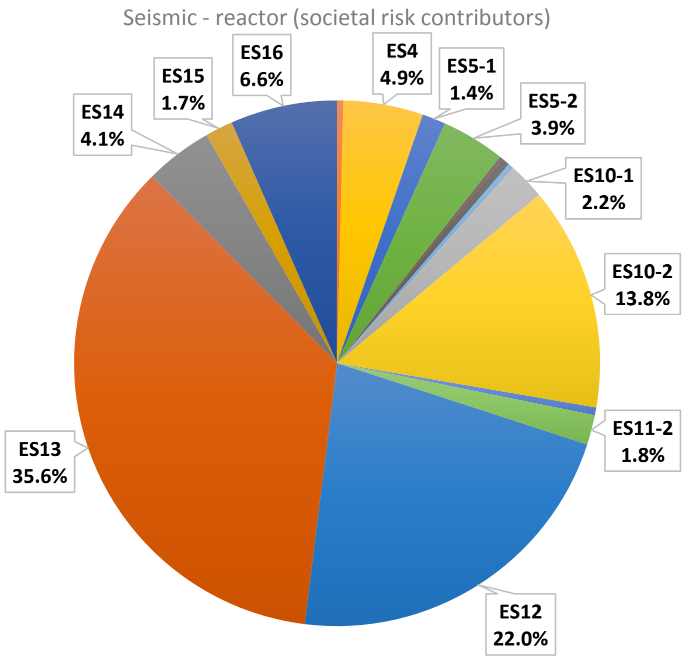


Figure 25.11.2-17 Contribution of Release Categories to the Frequency of Exceeding the Societal Risk Criterion for External Hazards: Seismic Event Affecting Reactor at power Leading to Fuel Melt

(b) Contribution from seismic event affecting the SFP leading to fuel melt

Table 25.11.2-18 presents the assessment against Target 9 for the External Hazards: seismic event affecting the SFP, based on the release category frequencies of Table 25.11.2-8 and the predicted societal consequences presented in Section 25.9. The release categories contributing to this frequency are shown in Figure 25.11.2-18.

The seismic event affecting the SFP contributes 3.86E-07 /y to the frequency of exceeding the Target 9 threshold with minimal protective actions. This is equivalent to 385 percent of the BSO or a factor of 3.9 above the BSO. This is equivalent to 3.9 percent of the BSL.

- Release category SBO-SFP02 is dominant and contributes 2.22E-07 /y (57.5 percent of the total frequency). This is above the BSO.
- A further three release categories contribute an additional 1.21E-07 /y or 31.3 percent of the total:
  - Release category SBO-SFP_LOCA02 at 8.10E-08 /y (21.0 percent of the total frequency),
  - Release category SBO-SFP03 at 2.82E-08 /y (7.3 percent of the total frequency), and
  - Release category SDDS-SFP02 at 1.14E-08 /y (3.0 percent of the total frequency).

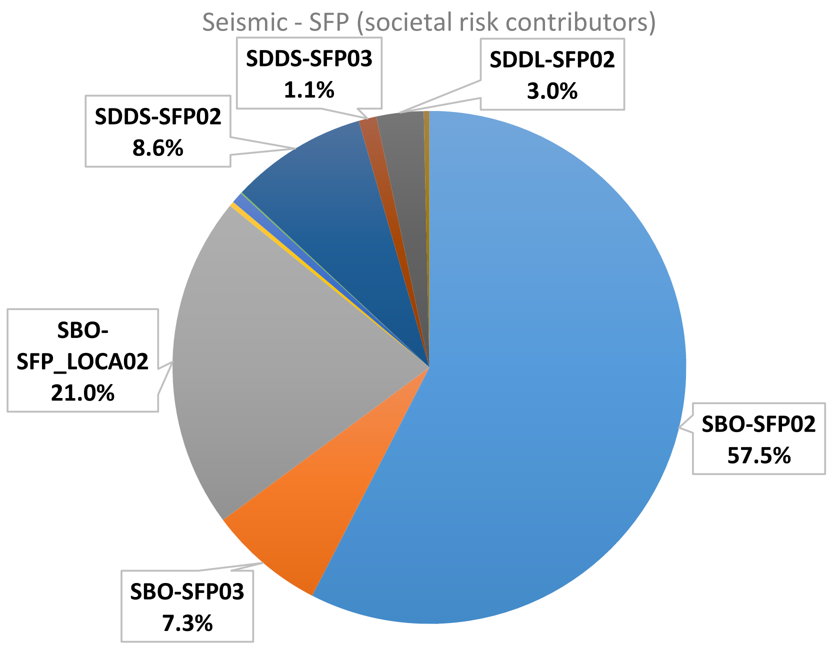


Figure 25.11.2-18 Contribution of Release Categories to the Frequency of Exceeding the Societal Risk Criterion for External Hazards: Seismic Event Affecting the SFP Leading to Fuel Melt

25.11.2.9 Uncertainty and Sensitivity Analysis

(1) Uncertainty Analysis

Uncertainty analysis has been performed using the Monte Carlo sampling method which generated a probability density function and a cumulative probability function for CDF and LRF. The uncertainty distribution and error factors have been captured in the type code (TC) table of the modified Level 1/Level 2 PSA database. In particular, the alpha-factor method has been used to quantify CCFs in the UK ABWR PSA and is preferred as it enables more accurate modelling of uncertainty in the CCF parameters.

The mean values are CDF of 8.0E-08 /y and LRF of 6.3E-07/y for the reactor. The mean values generated based on the sample size are FDF of 5.5E-07 /y and LRF of 5.0E-07 /y for the SFP.

In addition, uncertainty analysis for each release category has been performed as shown in Table 25.11.2-19 and Table 25.11.2-20.

The errors factors result from the uncertainty analyses were larger than those of internal events uncertainty analyses. This is because the seismic PSA includes uncertainty from seismic fragility, seismic hazard and random failure probabilities. The seismic fragility and the hazard curve have large variances

The mean values of CDF, FDF and LRF from UNCERT are larger than point estimate values. The reasons are as follows.

- As mentioned above, the seismic fragility and the hazard curve have large variances, which would cause some excessively high frequencies or probabilities during the Monte-Carlo simulations.
- Seismic PSA uncertainty analyses included the effect of SOKC (State of Knowledge Correlation) as in internal events PSA. Especially, the following combinations of basic events have a large SOKC effect because a number of cutsets include these combinations.
  - BBG-1 and BBG-2 (Loss of backup building generators)
  - H11-OLU-US-___-T1P2, H11-OLU-US-___-T2P2, H11-OLU-US-___-T3P2 (Output Login Unit undetectable loss of function)

Category	Mean	5 %	Median	95 %
CDF of reactor at power	8.84E-07	3.84E-08	3.77E-07	3.20E-06

Sample Size: 100,000

Category	Mean	5 %	Median	95 %
LRF of reactor at power	6.53E-07	2.35E-08	2.49E-07	2. 43E-06

Sample Size: 100,000

Category	Mean	5 %	Median	95 %
FDF of SFP	5.63E-07	2.47E-08	1.98E-07	2.15E-06

Sample Size: 100,000

Category	Mean	5 %	Median	95 %
LRF of SFP	5.14E-07	1.58E-08	1.65E-07	2.01E-06

Sample Size: 100,000

Table 25.11.2-19 Uncertainty Analyses for Reactor at power Release Category Frequencies

No.	Release Category	Mean	5 %	Median	95 %
1	Containment Leakage (KV and KP_I, II and III)	4.06E-08	6.29E-11	6.10E-09	1.84E-07
2	Containment Venting (VV and VP_I, II and III)	7.64E-09	9.32E-12	6.70E-10	2.91E-08
3	Filtered Containment Venting (FVV and FVP_I, II and III)	1.47E-07	1.80E-09	4.55E-08	5.96E-07
4	Early Containment Failure (ALL_IV_except_BYPASS)	6.21E-08	7.55E-12	4.63E-09	2.38E-07
5-1	Late Containment Failure (OP/OT_I)	2.91E-08	8.85E-11	5.34E-09	1.18E-07
5-2	Late Containment Failure (OP/OT_II)	4.40E-08	1.14E-10	6.35E-09	1.76E-07
5-3	Late Containment Failure (ALL_V_except_BYPASS)	2.38E-10	3.38E-16	1.88E-12	6.54E-10
5-4	Late Containment Failure (ALL_VI_except_BYPASS)	1.03E-09	1.90E-13	2.84E-11	3.45E-09
6	Late Containment Failure (OP/OT_PS_I, II and III)	5.31E-09	1.30E-11	7.58E-10	2.31E-08
7-1	In-vessel Fuel-Coolant Interaction (RE_I)	1.07E-11	2.94E-15	4.40E-13	2.95E-11
7-2	In-vessel Fuel-Coolant Interaction (RE_II)	7.28E-12	1.43E-15	2.49E-13	2.08E-11
8-1	Ex-vessel Fuel-Coolant Interaction (PE_I)	1.46E-10	5.08E-14	6.50E-12	4.26E-10
8-2	Ex-vessel Fuel-Coolant Interaction (PE_II)	3.34E-09	3.36E-12	2.41E-10	1.11E-08
9	Direct Containment Heating (DCH_II)	1.67E-10	8.69E-15	3.02E-12	4.61E-10
10-1	PCV Isolation Failure (PCI_I)	7.32E-08	1.57E-10	1.18E-08	3.01E-07
10-2	PCV Isolation Failure (PCI_II)	2.49E-07	2.67E-09	6.04E-08	9.61E-07
11-1	Molten Core Concrete Interaction (CCI_I)	6.67E-09	8.65E-14	3.21E-10	2.79E-08
11-2	Molten Core Concrete Interaction (CCI_II)	1.91E-08	2.35E-13	8.01E-10	7.36E-08
12	RPV Rupture (ALL_IX_except BYPASS and RR and RR_LD_AE)	2.79E-07	2.21E-09	7.30E-08	1.10E-06
13	Containment Bypass (ALL VII and BYPASS_except III)	5.60E-07	1.37E-08	1.96E-07	2.11E-06
14	S/P Bypass (ALL_VIII_except_BYPASS)	4.35E-08	4.03E-12	2.37E-09	1.73E-07
15	Direct Debris Interaction (DDI_II)	5.42E-08	4.37E-11	4.60E-09	2.29E-07
16	Long term SBO (Containment Failure w/o Spray _III)	6.74E-08	1.43E-10	9.85E-09	2.75E-07

Table 25.11.2-20 Uncertainty Analysis for SFP Release Category Frequency

No.	Release Category	Mean	5 %	Median	95 %
1	SDDL-SFP02	1.66E-08	1.20E-10	3.95E-09	6.52E-08
2	SDDS-SFP02	1.10E-07	4.24E-10	1.97E-08	4.43E-07
3	SBO-SFP02	3.91E-07	1.00E-08	1.18E-07	1.55E-06
4	SDDL-SFP03	1. 85E-09	1.24E-11	4.24E-10	7.08E-09
5	SDDS-SFP03	1.18E-08	4.60E-11	2.10E-09	4.69E-08
6	SBO-SFP03	4.55E-08	1.08E-09	1.30E-08	1.81E-07
7	SBO-SFP_LOCA02	1.57E-07	1.04E-09	3.58E-08	6.37E-07
8	SBO-SFP_LOCA03	8.83E-10	4.25E-12	1.79E-10	3.56E-09
9	SBO-SFP_LOCA04	1.21E-09	2.10E-15	4.04E-12	2.59E-09
10	SBO-SFP_LOCA05	8.35E-11	7.86E-18	2.57E-13	1.74E-10

(2) Sensitivity Analysis

Because of the substantial uncertainties associated with the quantitative seismic risk analyses, it is useful to gain insights on the results by examining variations in the risk estimates as a function of key conditions or assumptions.

In order to fully understand the “importance” of individual components and modelling inputs, it is appropriate to perform a series of sensitivity cases where these model inputs are varied. Variations are based on changes in assumptions, plant operating configurations, operator performance, equipment performance and etc. The results and insights from the sensitivity cases provide a basis for the identification of risk significant areas.

The sensitivity analyses included the following cases, which deemed to be significant:

- Uncorrelated failures of system components on both the front line and major support systems
- Impact of modelling the Seismic Category 3 equipment in Seismic SFP PSA
- Impact of sloshing in Seismic SFP PSA
- Impact of the PIF increase with different thresholds, which includes the HEP for FLSR
- Impact of the success branches

25.11.2.10 Insights from the Seismic Assessment

Reactor

- The largest contribution to CDF and LRF for reactor at power Seismic PSA Level 1 and Level 2 seismic PSA comes from 0.8 to 0.9 g PGA (Peak Ground Acceleration).
- In the Level 1 seismic PSA, at low ground acceleration below 0.6 g, SBO sequences are dominant; most of which are caused by EDG support system failures such as RSW pumps and DG Light Oil Tanks, whose seismic capacities are small. Above 0.6 g, the contribution of LOCA initiating scenarios gets larger with increasing seismic hazard level.
- In the Level 2 PSA, the contribution of TB sequences is large where the seismic hazard level is below 0.4 g. In high hazard levels above 0.4 g, S3E and AE sequences represent a large portion of LRF. Most of S3E scenarios are originated in the BOC sequence at the CUW line in the Level 1 PSA and this scenario has no mitigation in Level 2 PSA event tree. AE sequences are originated from the group of LOCA inside containment scenarios.

SFP

- The largest contribution to FDF and LRF for the SFP at Power Level 1 and Level 2 PSA comes from the 0.8 to 0.9 g (PGA). The sudden increase of FDF from 0.2 to 0.3 g to 0.3 to .4 g is due to the sloshing of the SFP water, leading to the loss of FPC.
- In the SFP Level 1 PSA, at low ground acceleration level below a specific value, damage sequences are dominated by the scenarios in LOOP conditions. In this scenario, the restart of FPC is credited since there is still sufficient water in SFP. At the seismic hazard above the specific value, FPC cannot be used due to SFP water sloshing leading to low water level in skimmer surge tank. In the 0.3 to 0.5 g seismic hazard interval, although Class 1 AC power is available in many cases, no make-up function can be credited by seismic category 3 system such as MUWC. Make up only by FLSS or FLSR can be credited in the seismic hazard interval. At seismic hazards above 0.5 g, the sequences in SBO condition dominate the total scenarios followed by SFP small leak sequence.
- In the SFP Level 2 PSA, Boil-off scenarios represent most of the LRF sequences through all bands of seismic hazard. The boil off scenarios can be classified into two groups; one caused by the failure of FPC and another caused by reactor BOC. With increasing seismic hazard level, the contribution of scenarios with reactor BOC becomes large.

Shutdown

- Although the quantification includes conservatism, the characteristics of risk contribution for each shutdown POS is similar to that of the internal events PSA. Application of the one fails-all fail rule degrades the effect of redundant systems and then the risk contribution of POS C (which has an especially large contribution in internal events) is reduced. However, the risk contribution of POS S and POS D (which are insignificant risk contributors in the internal events PSA) becomes large in the seismic PSA. Redundant systems decreased the risk in internal events; however, the one fails-all fail rule degrades the effect of the redundant systems for the seismic PSA.
- The CDF of the seismic shutdown PSA for the reactor is 4.2E-08 /y. This is about 6 percent of that of the at power condition. The percentage of shutdown risk contribution is reasonable because of duration of the shutdown outage, the absence of reactivity control accidents, the absence of seismic specific initiating events in shutdown, and the relatively low decay heat level compared to the at power condition.

25.11.2.11 Key Assumptions and Limitations

Assumptions in the Seismic PSA were made in the development phase. They relate to each aspects of PSA, such as the identification of initiating events, the definition of success criteria, reliability data selection, containment performance, accident progression and so on. In addition, when the level of plant requirements specification limits the detail of the PSA, then appropriate assumptions were developed to allow completion of the PSA.

Starting from assumptions, sensitivity analyses have been performed. Among assumptions in the Seismic PSA, key assumptions, which have comparatively large impact on the result, have been listed from the result of sensitivity analyse.

The key assumptions considered in the Seismic PSA are listed below.

- The “one fails-all fail” rule assumes that if one component fails due to the seismic event, then all components of that same type for that system in the same elevation and the same building fail as well, thus creating a common cause failure event.
- The most fragile components for the offsite grid and distribution system are "ceramic insulators" in the switchyard. This representative equipment is considered as the cause of LOOP in this seismic PSA. If this most fragile component does not fail by seismic impact, no initiating event is assumed in the reactor.
- Seismic HFE is evaluated based on IE PSA model by multiplying a stress factor. It is assumed that the stress factor increases under large earthquake giving higher stress.
- Sloshing of SFP water increases as a function of ground acceleration. With hazard level higher than a specific value, spill out of water from FPC skimmer surge tank as well as SFP is assumed to give "low-water level" where FPC pump automatically stops, and thus FPC cannot be credited.

25.11.3 Other External Hazard PSA

25.11.3.1 Assessment of Tornado Missiles

This section describes the tornado missile assessment to identify potential vulnerabilities in the UK ABWR. The PSA in this section is used to demonstrate that the generic design meets the NSEDP's numerical targets (equivalent to Targets 7, 8, and 9 of the SAP) for the GDA and to inform the design organizations of potential vulnerabilities to guide future activities, including the support of ALARP demonstration.

In order to identify the external hazards and internal hazards that are risk significant and necessary to consider in the assessment of risk for the GDA of the UK ABWR, the screening and prioritisation analyses of external hazard (EH) were presented in Section 25.11.1.

(1) Scope

The scope of this PSA was the same as that of the turbine missiles assessment as described in Section 25.10.5.1(1). The international TORRO Tornado Intensity Scale is shown in Table 25.11.3-1. For the PSA, two cases were developed with the dedicated hazard frequencies as shown in Table 25.11.3-2.

In this study, the annual exceedance frequency of T5 or greater tornado is presented as T5 frequency only. However, it was concluded that the risk from missiles induced by potential T6 or greater tornados was already captured by the conservative assignment of consequences from missiles induced by T3 or greater tornados.

(2) Methodology

The CDF and LRF of each scenario are assessed by multiplying the associated Hazard-Induced Initiating Event (IE) frequency by the CCDP and CLRP, respectively. This section describes the methodologies to derive the CCDPs and CLRPs. The postulated impacts from tornado missiles included Loss of Offsite Power (LOOP), loss of CST and loss of Light Oil Tank (LOT) (for the Backup Building Generators: BBGs). The CCDPs and CLRPs were calculated by the following process.

(1) Prepare cutset files for CDF and LRF which include all the cutsets for the specific IE TEW4 (Weather-related LOOP > 14 h), by merging the relevant cutset files generated by the sequence by sequence quantification for the Level 1 PSA [Ref 25-8] and the Level 2 PSA [Ref 25-74] with a truncation of 5.0E-17 /y, and deleting all the cutsets which include other IEs.

(2) Determine the surrogate basic events to represent the assumed conditions.

(3) Set the IE basic event to TRUE in order to derive the CCDP and CLRP in the cutset files.

(4) Set the basic event(s) for the determined basic events(s) to TRUE in the cutset files.

- Case I (LOOP only): None
- Case II (LOOP with loss of CST and LOT):

  CCF-BBG, BBG-1, BBG-2 (to represent loss of LOT)

  P13-AOV-CC-__-F013 (to represent loss of makeup to CST from PWST)

(5) Minimise (subsume) the cutset file.

(6) If a Level 1 PSA event tree heading is predetermined to fail due to the treatment in step 4), set to FALSE the Level 1 PSA sequence flag events for the sequences which include success of that heading, in order to avoid unintended non-minimal cutsets.

(3) Assessment

The CCDPs and CLRPs of tornado missile were assessed for two hazard levels, and the CDFs and LRFs were evaluated.

Case I

LOOP is initiated by a T0, T1 or T2 tornado which is shown in Table 25.11.3-1 [Ref 25-116]. Offsite power does not recover. There is no other impact on the SSCs or operator actions.

Case II

LOOP is initiated by a T3, T4 or T5 tornado which is shown in Table 25.11.3-1. Offsite power does not recover. LOT is unavailable. CST is unavailable but RCIC and HPCF are re-aligned to S/P suction mode considering the random failures which disable the re-alignment.

(a) Evaluation of CCDPs and CLRPs

Case I: Tornado induced un-recoverable LOOP

The calculated CCDP and CLRP are 5.3E-06 and 1.2E-06, respectively.

Case II: Tornado induced un-recoverable LOOP with loss of CST and LOT

The calculated CCDP and CLRP are 5.6E-04 and 2.6E-04, respectively.

(b) Evaluation of CDFs and LRFs

Based on the frequency estimated in Table 25.11.3-2 and the associated CCDP and CLRP, the CDF and LRF from each hazard level are calculated. Table 25.11.3-3 summarises the results.

Missiles generated by T0, T1 and T2 tornadoes (frequency = 9.4E-10 /y + 1.7E-08 /y + 1.2E-07 /y = 1.4E-07 /y) does not impact any systems credited for the internal events PSA but is assumed to cause un-recovered LOOP. The CCDP and CLRP for this situation are evaluated as 5.3E-06 and 1.2E-06, respectively. The CDF and LRF are calculated as 7.4E-13 /y (1.4E-07 /y times 5.3E-06) and 1.7E-13 /y (1.4E-07 /y times 1.2E-06), respectively.

Missiles generated by T3, T4 and T5 tornadoes (annual exceedance frequency of T3 = 9.3E-07 /y) impact the CST and LOT as well as causing an un-recovered LOOP. The CCDP and CLRP for this situation are evaluated as 5.6E-04 and 2.6E-04, respectively. The CDF and LRF are calculated as 5.2E-10 /y (9.3E-07 /y times 5.6E-04) and 2.4E-10 (9.3E-07 /y times 2.6E-04), respectively.

The frequencies of T6 or stronger tornadoes are estimated to be negligibly small.

The total CDF and LRF are three and two orders of magnitude lower than the CDF and LRF from internal initiating events at power, respectively (presented in Section 25.7).

Table 25.11.3-1 International TORRO Tornado Intensity Scale

TORRO scale	Potential damage
T0	Loose light litter rose from ground-level in spirals. Tents, marquees seriously disturbed; most exposed tiles, slates on roofs dislodged. Twigs snapped; trail visible through crops.
T1	Deckchairs, small plants, heavy litter becomes airborne; minor damage to sheds. More serious dislodging of tiles, slates, chimney pots. Wooden fences flattened. Slight damage to hedges and trees.
T2	Heavy mobile homes displaced, light caravans blown over, garden sheds destroyed, garage roofs torn away, much damage to tiled roofs and chimney stacks. General damage to trees, some big branches twisted or snapped off, small trees uprooted.
T3	Mobile homes overturned / badly damaged; light caravans destroyed; garages and weak outbuildings destroyed; house roof timbers considerably exposed. Some of the bigger trees snapped or uprooted.
T4	Motor cars levitated. Mobile homes airborne / destroyed; sheds airborne for considerable distances; entire roofs removed from some houses; roof timbers of stronger brick or stone houses completely exposed; gable ends torn away. Numerous trees uprooted or snapped.
T5	Heavy motor vehicles levitated; more serious building damage than for T4, yet house walls usually remaining; the oldest, weakest buildings may collapse completely.
T6	Strongly built houses lose entire roofs and perhaps also a wall; windows broken on skyscrapers, more of the less-strong buildings collapse.
T7	Wooden-frame houses wholly demolished; some walls of stone or brick houses beaten down or collapse; skyscrapers twisted; steel-framed warehouse-type constructions may buckle slightly. Locomotives thrown over. Noticeable debarking of trees by flying debris.
T8	Motor cars hurled great distances. Wooden-framed houses and their contents dispersed over long distances; stone or brick houses irreparably damaged; skyscrapers badly twisted and may show a visible lean to one side; shallowly anchored high rises may be toppled; other steel-framed buildings buckled
T9	Many steel-framed buildings badly damaged; skyscrapers toppled; locomotives or trains hurled some distances. Complete debarking of any standing tree-trunks.
T10	Entire frame houses and similar buildings lifted bodily or completely from foundations and carried a long or large distance to disintegrate. Steel-reinforced concrete buildings may be severely damaged or almost obliterated.

Table 25.11.3-2 Summary of Postulated Condition of Tornado for PSA with Occurrence Frequency

Case	Postulated condition	TORRO scale	IE Frequency (/y)
I	LOOP due to tornado	T0, T1, T2	1.4E-07
II	Loss of LOT for B/B, CST tank with LOOP	T3, T4, T5	9.3E-07

Table 25.11.3-3 Summary of Tornado Missile Risk Assessment

Case	Target of tornado missile	IE Frequency (/y)	CCDP	CDF (/y)	Contribution to CDF	CLRP	LRF (/y)	Contribution to LRF
I	Offsite power	1.4E-07	5.3E-06	7.4E-13	0.1 %	1.2E-06	1.7E-13	0.1 %
II	Offsite power, LOT, CST	9.3E-07	5.6E-04	5.2E-10	99.9 %	2.6E-04	2.4E-10	99.9 %
-	-	-	-	Total: 5.2E-10	Total: 100 %	-	Total: 2.4E-10	Total: 100 %

(4) Uncertainty and Sensitivity Analysis

The total CDF and LRF for tornado missile hazards are calculated as 5.2E-10 /y and 2.4E-10 /y, respectively, by using the simplified method. Since it is confirmed that these are three orders magnitude lower than the CDF and LRF from the internal initiating events at power, the uncertainty and sensitivity analyses was not performed in the GDA PSA.

(5) Insights from Assessment

The quantification results provide the following insights:

- Summation of these CDFs is 5.2E-10 /y. Summation of these LRFs is 2.4E-10 /y.
- Contributions from the missiles by T0, T1 and T2 tornadoes to the CDF and LRF are negligible because of the smaller IE frequency and more limited impact (LOOP only).

(6) Key Assumptions and Study Limitations

Assumptions in the tornado missiles PSA were made in the development phase. They relate to each aspect of PSA, such as the identification of initiating events, the definition of success criteria, reliability data selection, containment performance, accident progression and so on. In addition, when the level of plant requirements specification limits the detail of the PSA, then appropriate assumptions were developed to allow completion of the PSA.

The key assumptions considered in the tornado missiles PSA are listed below.

- Offsite power does not recover given a tornado missile induced LOOP.
- CST and LOT are conservatively assumed impacted by a missile induced by T3 tornado.
- The Human Error Probabilities (HEPs) used for the IEAP Level 1 PSA and Level 2 PSA are applicable to the tornado missile risk analysis for the operator actions performed in the Main Control Room (MCR). This is because the tornado missiles do not impact the civil structure of the Control Building (CB).
- The HEPs for the long term actions performed outside the MCR, credited in the tornado missile risk analysis, are assumed to not be impacted by the tornado missile.
- FLSR (Mobile Injection Facility) Unavailability is not impacted by a turbine missile because FLSR is operated in the yard with the time available of 8 hours or more.

- The impact on the cutsets which are below 1.0E-14 /y is not captured. This is a limitation of this simplified approach.
- In addition, it is qualitatively argued that the risk from tornado missiles during shutdown states and on SFP (both at power and shutdown) is insignificant. Since the initiating event frequency is larger and fewer systems are credible for the Case II (missiles generated by T3, T4 and T5), Case II is specifically discussed. During shutdown, the total initiating event frequency (per calendar year) is already smaller than the LRF from internal initiating events at power, during shutdown and SFP. Given multiple heat removal/makeup systems are credible, the LRF of reactor and SFP from tornado missiles during shutdown is deemed insignificant. For POS F (at power), the initiating event frequency is an order of magnitude higher than the LRF from internal initiating events at power and SFP. However, the LRF of SFP from tornado missiles at power is also deemed insignificant since two FPC divisions and FLSR are credible. In addition to the above discussions, large time available during shutdown states and for SFP makeup (given a LOOP event) would make the successful recoveries (i.e., repair of randomly failed components and injection) more likely than the case of reactor at power. Overall, the risk from tornado missiles on the reactor during shutdown and SFP (including at power and shutdown states) is deemed insignificant compared to the risk of reactor from internal events at power and risk of SFP from internal events.

25.11.3.2 Assessment of Accidental Aircraft Impact

This section describes the PSA for the Accidental Aircraft Impact to identify potential vulnerabilities in the UK ABWR. The PSA in this section is used to demonstrate that the generic design meets the NSEDP's numerical targets (equivalent to Targets 7, 8, and 9 of the SAP) for the GDA and to inform the design organizations of potential vulnerabilities to guide future activities, including the support of ALARP demonstration.

In order to identify the external hazards and internal hazards that are risk significant and necessary to consider in the assessment of risk for the GDA of the UK ABWR, the screening and prioritisation analyses of external hazard (EH) were presented in Section 25.11.1.

(1) Scope

The scope of this PSA is the same as that of the tornado missiles assessment as described in Section 25.10.5.1(1). The hazard frequency is shown in [Ref 25-1].

(2) Methodology

CDF and LRF of each scenario are assessed by multiplying the associated Hazard-Induced Initiating Event (IE) frequency by the CCDP and CLRP, respectively. This section describes the methodologies to derive the CCDPs and CLRPs.

The CCDPs and CLRPs are calculated by the following process. This is based on the process used for the turbine missile assessment (see Section 25.10.5). The differences from the process used for the turbine missile assessment are explained.

(i) Prepare cutset files for CDF and LRF which include all the cutsets for the specific IE TEW4 (Weather-related LOOP > 14h). This step is identical to that for the turbine missile assessment.

(ii) Set the IE basic event to TRUE in order to derive CCDP and CLRP in the cutset file. This step is identical to that for the turbine missile assessment (Section 25.10.5).

(iii) Set all of the Level 1 PSA sequence flag events and PDS flag events to TRUE. This step is performed in this study while not performed in the turbine building assessment. That is because the whole building (given impact to C/B or Hx/B) or two divisions (given impact on R/B) are assumed to be impacted in this study. Non-minimal cutsets due to the existence of the sequence flag files and PDS flag files could result in significant overestimation of the CCDP and CLRP (up to twice) given a wide range of basic events are set to TRUE.

(iv) Set the basic events mapped to the specific building/division to TRUE in the cutset file. Use the same component to building/division mapping as for the turbine missile assessment (Section 25.10.5). One difference from the turbine missile assessment is that the basic events of check valves or manual valves of FLSS (E71) located in RB are not made TRUE given loss of the Reactor Building.

(v) Additionally, set the selected events to TRUE in the case of an accidental aircraft impact on the C/B or Hx/B.

This step is unique to this study compared to the turbine missile assessment (Section 25.10.5). The objective is to capture the impacts on the truncated combinations of basic events. For example, a combination of loss of four Class 1 DC buses was truncated in the base IEAP PSA but this could occur given loss of all C/B divisions. To capture this, all of the CCF events of the Class 1 DC buses are set to TRUE.

(vi) Set the selected fail-safe type basic events (related to RPS scram) to FALSE in the case of an accidental aircraft impact on C/B.

This step is unique to this study compared to the turbine missile assessment (Section 25.10.5). The objective is to consider the fail-safe feature of the RPS scram function, except for the mechanical failure of control rods, given a loss of all C/B divisions

(vii) Minimise (subsume) the cutset file.

(3) Assessment

The CCDPs and CLRPs are assessed for the cases with an accidental aircraft impact on one of the following buildings. Then CDFs and LRFs are evaluated by multiplying the accidental aircraft impact frequencies with the associated CCDPs and CLRPs.

<u>Case I</u>

An accidental aircraft hits the R/B and impacts two of four divisions in the R/B. CCDP and CLRP given a loss of all combinations of two divisions are calculated and the highest CCDP and CLRP are used for the CDF and LRF evaluations, respectively.

<u>Case II</u>

An accidental aircraft hits the C/B and impacts all of the SSCs inside the C/B. CCDP and CLRP given this situation are calculated and used for the CDF and LRF evaluations, respectively.

<u>Case III</u>

An accidental aircraft hits the Hx/B and impacts all of the SSCs inside the Hx/B. CCDP and CLRP given this situation are calculated and used for the CDF and LRF evaluations, respectively.

(a) Evaluation of CCDPs and CLRPs

<u>Case I:</u> Accidental Aircraft Impact on R/B

The CCDPs and CLRPs given loss of two divisions of the R/B are calculated by the process introduced in Section 25.11.3.2 (2).

Loss of R/B Div. I and II:	CCDP = 7.3E-04	CLRP = 2.4E-04
Loss of R/B Div. I and III:	CCDP = 1.7E-03	CLRP = 3.5E-04
Loss of R/B Div. I and IV:	CCDP = 1.1E-04	CLRP = 1.7E-05
Loss of R/B Div. II and III:	CCDP = 5.4E-04	CLRP = 2.2E-04
Loss of R/B Div. II and IV:	CCDP = 1.1E-04	CLRP = 1.4E-05
Loss of R/B Div. III and IV:	CCDP = 9.0E-05	CLRP = 1.9E-05

Case II: Accidental Aircraft Impact on C/B

The CCDP and CLRP given a loss of the overall C/B are calculated as 1.1E-02 and 6.4E-03, respectively.

Case III: Accidental Aircraft Impact on Hx/B

The CCDP and CLRP given a loss of the overall Hx/B are calculated as 4.7E-03 and 3.4E-03, respectively.

(b) Evaluation of CDFs and LRFs

Based on the frequencies estimated in [Ref 25-1] and the associated CCDPs and CLRPs evaluated in Section 25.11.3.2(3) (a), the CDF and LRF from each target case were calculated.

Summation of these CDFs is 7.9E-10 /y. This is three orders of magnitude lower than the CDF from internal initiating events at power (presented in Section 25.7).

Summation of these LRFs is 4.6E-10 /y. This is two orders of magnitude lower than the LRF from internal initiating events at power (presented in Section 25.7).

(4) Uncertainty and Sensitivity Analysis

The total CDF and LRF for turbine missile hazards are calculated as 7.9E-10 /y and 4.6E-10 /y, respectively, by using the simplified method. Since it is confirmed that these are three orders magnitude lower than the CDF and LRF from the internal initiating events at power, the uncertainty and sensitivity analyses was not performed in the GDA PSA.

(5) Insights from Assessment

The quantification results provide the following insights:

- Summation of these CDFs is 7.9E-10 /y. Summation of these LRFs is 4.6E-10 /y. Contribution form the Hx/B case is the highest (CDF: 47 percent, LRF: 59 percent) due to the second highest IE frequency and the highest CCDP resulting from the loss of all HPCF and RHR divisions.

In addition, it is qualitatively argued that the frequency of accidental aircraft impacts on the reactor during shutdown states and on SFP (both at power and shutdown) is insignificant. Structural failure of SFP itself or spent fuels due to direct impact of aircraft or overhead crane drop are negligible risk impact since it is prevented by design. As such, LOOP with loss of specific building/division is considered as the risk contributor. During shutdown, the total initiating event frequency (per calendar year) is already smaller than the LRF from internal initiating events at power, during shutdown and SFP. Given multiple heat removal/makeup systems are credible, the LRF of reactor and SFP from accidental aircraft impact during shutdown is deemed insignificant. For POS F (at power), the initiating event frequency for each building is in the same order as the LRF from internal initiating events at power and SFP. However, the LRF of SFP from accidental aircraft impact at power is also deemed insignificant since multiple makeup systems are credible. In addition to the above discussions, large time available during shutdown states and for SFP makeup (given a LOOP event) would make the successful recoveries (i.e., repair of randomly failed components and injection) more likely than the case of reactor at power. Overall, the risk from tornado missiles on the reactor during shutdown and SFP (including at power and shutdown states) is deemed insignificant compared to the risk of reactor from internal events at power and risk of SFP from internal events.

(6) Key Assumptions and Limitations

Assumptions in the Accidental Aircraft Impact assessment were made in the development phase. They relate to each aspect of PSA, such as the identification of initiating events, the definition of success criteria, reliability data selection, containment performance, accident progression and so on. In addition, when the level of plant requirements specification limits the detail of the PSA, then appropriate assumptions were developed to allow completion of the PSA.

The key assumptions considered in the Accidental Aircraft Impact assessment are listed below.

- Non-recoverable LOOP is assumed to always occur given occurrence of an accidental aircraft impact to the Reactor Building (R/B), Control Building (C/B) or Heat Exchanger Building (Hx/B), for the evaluations of bounding CCDPs and CLRPs. Therefore, SSCs for the systems/functions credited in the CCDP and CLRP assessments do not include the power conversion system, feedwater system, condensate system and their supporting systems. This is already captured by using the surrogate initiating event (Weather-related LOOP > 14 h) since the related headings are excluded from the event tree structure and the offsite power is disabled for all the sequences by the flag settings.
- Two divisional areas of the R/B are lost given an accidental aircraft impact on the R/B but other buildings/structures are not impacted.
- For the screened-in C/B scenarios, all divisional areas of the C/B are assumed to be lost given an accidental aircraft impact on the C/B but other buildings/structures are not impacted.
- For the screened-in Hx/B scenarios, all of the divisional areas of the Hx/B are assumed to be lost given an accidental aircraft impact on the Hx/B but other buildings/structures are not impacted.
- Potential fire-induced spurious events are not induced by an accidental aircraft impact (random spurious events are considered). This is a limitation of the current simplified approach and involves potential non-conservative bias.
- The same HEPs as used for the IEAP Level 1 PSA and Level 2 PSA are applicable to an accidental aircraft impact on the R/B or Hx/B for the operator actions performed in the MCR. This is because the MCR is assumed not to be impacted.
- The HEPs (credited in this study) for the long term actions performed at the Back-up Building (B/B) control room are not affected by accidental aircraft impact on the R/B, C/B or Hx/B because the B/B is assumed to be not impacted in these scenarios.
- The HEPs (credited in this study) performed inside the C/B but outside the MCR are not impacted by accidental aircraft impact on the R/B or Hx/B because the C/B is assumed to be not impacted in these scenarios.
- When the CDF and LRF from an accidental aircraft impact on the R/B are evaluated, the impact frequency on the R/B (given in Appendix G) is used as the initiating event frequency. Loss of any combination of two divisions is analysed regardless of the relative locations of the divisional areas.

Then, the highest CCDP and CLRP are representatively used among all the possible combinations of the two R/B divisions impacted.

- The impact on the cutsets which are below 1.0E-14 /y is not captured. This is a limitation of this simplified approach.

25.11.3.3 Assessment of External Flooding and Biological Fouling

This section describes the PSA for the External Flooding and Biological Fouling to identify potential risk in the UK ABWR. Hazard assessment of External Flooding and Water based Biological Fouling may not be available in the GDA time frame. Under this situation, CDFs for these external hazards are semi-quantitatively assessed to identify potential vulnerabilities in the UK ABWR as a sensitivity analysis.

In order to identify the external hazards and internal hazards that are risk significant and necessary to consider in the assessment of risk for the GDA of the UK ABWR, the screening and prioritisation analyses of external hazard (EH) were presented in Section 25.11.1.

(1) Scope

The scope of this PSA is the same as that of tornado missiles assessment as described in Section 25.10.5.1(1). The hazard frequencies related to external flooding and biological fouling depend on the site conditions and thus those are assumed as shown in Table 25.11.3-4 and Table 25.11.3-5.

Table 25.11.3-4 Assumption of External Flooding Frequency

ID	External Flood level for Wylfa point D	IE Frequency (/y)
I	12.9 m	1.0×10^{-4}
II	14.3 m	1.6×10^{-6}
III	14.5 m	6.3×10^{-7}

Table 25.11.3-5 Assumption of Intake Blockage Frequency by Water Based Biological Fouling

ID	Postulated IE Case	IE Frequency (/y)
I	The intake function when CW pump is lost	1.0×10^{-2}
II	The intake function when RSWs is lost	1.0×10^{-4}

(2) Methodology

The risk for External Flooding and Water based Biological Fouling is calculated by multiplying the estimated initiating frequency and CCDPs to calculate an estimated CDF. The CCDPs are calculated for Case 1 to Case 4 shown in Table 25.11.3-6, and the cutsets/importance is assessed in Section 25.11.3.3 (3). The CCDP given site flooded or sea water intake blockage is calculated in sections below by the following process:

(1) Prepare the cutset file generated by the one-top quantification in [Ref 25-8] with a truncation of 1.0E-14.

(2) Determine the surrogate basic events (including IE basic events) to represent the assumed condition.

(3) Set the basic event(s) for the determined basic events(s) to TRUE in the existing cutset file. The basic events, which are TRUED, are described in Section 25.4.1.

- Case 1: Manual Shutdown (CW pump trip),
- Case 2: Manual Shutdown, Loss of RSW/TSW,
- Case 3: Manual Shutdown with LOOP, Loss of FLSR, and
- Case 4: Manual Shutdown with LOOP, Loss of RSW/TSW, RCIC and FLSR.

(4) Set the basic event "PLANT-AVAILABILITY" to TRUE.

(5) Minimise (subsume) the cutset.

(6) Filter the cutset by the determined IE basic event and derive the CCDP.

Evaluation for each hazard, including discussion of acceptance criteria, is described as shown below.

External Flooding

For the External Flooding evaluation, the CCDPs of Cases 1, 3 and 4 are multiplied by the assumed external flooding frequencies. The insight on the LOOP is developed by comparing Case 1 and Case 3. The insight on the event of further damage to components in the Reactor Building is developed by comparing Case 3 and Case 4.

Biological Fouling

For the water based biological fouling evaluation, the CCDPs of Case 1 and Case 2 are multiplied by the assumed IE frequency. The insight on the intake blockage for CW and RSW is developed by comparing Case 1 and Case 2. Detailed analysis on the difference between Case 1 and Case 2 may provide insight for future design modification, such as Reserved UHS.

Table 25.11.3-6 Analysis Cases for Loss of Ultimate Heat Sink

Case	CW (PCS)	Offsite Power	RSW	TSW	RCIC	FLSS	FLSR	Containment Venting	Area
1	N/A	A	A	A	A	A	A	A	E, B
2	N/A	A	N/A	N/A	A	A	A	A	B
3	N/A	N/A	A	A	A	A	N/A	A	E
4	N/A	N/A	N/A	N/A	N/A	A	N/A	A	E

A: Available, N/A: Not Available

E: Used in the External Flooding, B: Used in the Biological Fouling

(3) Assessments

External Flooding

CCDPs for four representative flood levels and CCDPs of Cases 1, 3 and 4 are multiplied by the assumed external flooding frequencies. Analysis results are summarised in Table 25.11.3-7.

Table 25.11.3-7 External Flooding CDF by Simplified Approach

ID	Postulated IE Case	IE Frequency (/y)	CCDP (-)	Analysis Case	CDF (/y)
I	The site is flooded over nuclear island platform and the plant is administratively shutdown. The water does not reach transformer and the opening of R/B and H/B, components in the buildings. IE frequency is conservatively estimated at the level of Design Base flood.	1E-4	1.5E-9	Case 1	1.5E-13
II	The site is flooded over nuclear island platform and the plant is administratively shutdown. Water level does not reach the opening of Hx/B and R/B and active components in these buildings are available. FLSR is not available due to the flood on the yard. IE frequency is estimated at the level of 14.3 m – 14.5 m	9.7E-7*	5.2E-6	Case 3	5.0E-12
III	Water level reaches the opening of Hx/B and R/B and active components in these buildings in the lower grade are not available. FLSR is not available due to the flood on the yard. IE frequency is estimated at the level of 14.5 m.	6.3E-7	5.4E-3	Case 4	3.4E-09

(*) 9.7E-7 = (1.6E-6) – (6.3E-7)

Water Based Biological Fouling

CCDPs of Case 1 and Case 2 are multiplied by the assumed intake blockage frequencies. Calculated CDFs are shown in Table 25.11.3-8.

Table 25.11.3-8 Water Based Biological Fouling CDF by Simplified Approach

ID	Postulated IE Case	IE Frequency (/y)	CCDP (-)	Analysis Case	CDF (/y)
I	The intake function for CW pump is lost.	1E-2	1.5E-9	Case 1	1.5E-11
II	The intake function for CW, RSW and TSW is lost	1E-4	7.4E-4	Case 2	7.4E-8

(4) Uncertainty and Sensitivity Analysis

External flooding and biological fouling are assessed as a sensitivity analysis because Hazard assessment of External Flooding and Water based Biological Fouling may not be available in the GDA time frame. Therefore, uncertainty and sensitivity analysis for the assessment is not performed.

(5) Insights from Assessment

The quantification results of external flooding provide the following insights:

- Summation of these CDFs is 3.4E-9 /y. Major contributor in the external flooding level is over the opening in the R/B and Hx/B (ID III and over ID III), because the CCDP increase is smaller than the reduction of the flooding frequency between ID II (Case 3) and ID III (Case 4). On the other hand, ID I is the smallest contributor because multiple mitigation features are available and its CCDP is the smallest. According to these characteristics, protection of flooding to the R/B and Hx/B is an effective measure to reduce this risk.
- ID I case is that external flooding reaches between the Design Base Flood and the opening of the Reactor Building. The IE frequency (1E-4 /y), which is the level of the Design Base Flood, is conservatively used. Although the PCS is not available in this IE, FW, RCIC, HPCF, RHR FLSS and FLSR are available. The offsite power is also available. Since these multiple measures are available, the CCDP is 1.5E-9, and the estimated CDF is 1.5E-13 /y.
- ID II case is that external flooding reaches the level below the opening of the Reactor Building, Heat Exchanger Building and the transformer for the offsite power. The IE frequency, 1.6E-6 /y is estimated at the level of the 14.3 m. The IE frequency between 14.3 m and 14.5 m is 9.7E-7 /y. In this IE, RCIC, HPCF, RHR and FLSS are available. The offsite power and FLSR are not available due to the flood over the transformer and mobile connection area. Since these multiple measures are available, the CCDP is 5.2E-6. Then, the estimated CDF is 5.0E-12 /y.
- ID III case is that external flooding reaches the opening of the Reactor Building, Heat Exchanger Building and the transformer for the offsite power. The IE frequency (6.3E-7 /y) is conservatively estimated at the level of the 14.5 m. In this IE, FLSS is available under the loss of offsite power. The CCDP is 5.4E-3. Then, the estimated CDF is 3.4E-09 /y.

The quantification results of water based biological fouling provide the following insights:

- Summation of these CDFs is around 7E-8 /y. The major contributor in the intake blockage is the loss of RSW/TSW (ID II), because its CCDP is significantly larger than that in ID I (Case 1). According to this, the countermeasure to blockage of RSW intake or alternative measure of RSW is effective measure to reduce this risk.
- ID I case is that the intake for CW is blocked by water based biological fouling. The IE frequency (1E-2 /y) is estimated as the blockage by Design Base biological fouling. Although the PCS is not available in this IE, FW, RCIC, HPCF, RHR FLSS and FLSR are available. Offsite power is also available. Since these multiple measures are available, the CCDP is 1.5E-9, and the estimated CDF is 1.5E-11 /y.
- ID II case is that the intake for RSW/TSW is blocked by water based biological fouling. The IE frequency (1E-4 /y) is estimated considering connection to CW and effect of the screen. In this IE, RCIC, FLSS and FLSR are available. Offsite power is also available. Since these multiple measures are available, the CCDP is 7.4E-4, and the estimated CDF is 7.4E-8 /y.

(6) Key Assumption and Study Limitations

Assumptions in the External Flooding and Biological Fouling assessment were made in the development phase. They relate to each aspects of PSA, such as the identification of initiating events, the definition of success criteria, reliability data selection, containment performance, accident progression and so on. In addition, when the level of plant requirements specification limits the detail of the PSA, then appropriate assumptions were developed to allow completion of the PSA.

Starting from assumptions, sensitivity analyses have been performed. Key assumptions, which have a comparatively large impact on the results, have been listed from the result of sensitivity analyses.

The key assumptions considered in the External Flooding and Biological Fouling assessment are listed below.

- Sea water level by external flooding goes up with significant time margin after the detection of its symptom or its warning. As the sea water rises, there is significant time after the detection of the rise until the plant equipment is flooded. Thus, administrative operator actions for plant safe shutdown are available.
- The floors of the RSW pumps and TSW pumps are water sealed. Then, water comes into the Heat Exchanger building from the entrance above the ground level.
- The wall between the Reactor Building and the tunnel to the Backup Building is water sealed and the tunnel to the opening of the Backup Building is water-tight. If the wall seal or tightness is not achieved in the future design, equipment (e.g., piping, check valve) installed lower than the opening of the Backup Building is water resistant nature. Then, the equipment in the Backup Building is available, if the water level is under the opening of the Backup Building.
- In the event that the sea water level is going up, reactivity control completes by the control rod, before the sea water level reaches the nuclear island platform.
- The amount of water entering the Reactor Building is assumed to be smaller than the necessary amount for flooding the Class 2 transmitters, i.e., these Class 2 transmitters are assumed not to be impacted by external flooding.
- If plant monitoring parameters are not available due to the flood in the Reactor Building, FLSS water injection rate is controlled based on the decay heat level.
- Intake blockage by biological fouling is a slowly progressing phenomenon. Thus, administrative operator actions for plant safe shutdown are available.
- The intake blockage causes complete loss of CW or RSW/TSW. The loss of RSW/TSW has different characteristics from the loss of CW. That is, the loss of CW does not always correlate to the loss of RSW/TSW due to the different required capacity of intake.
- If the site is flooded over nuclear island platform, the FLSR (having location diversity with the nuclear island platform) is assumed to be available for a minimum of 24 hours after the Initiating Event.
- FLSR is available to at least a water level of 0.3 m above ground level.
- The impact on the cutsets which are below 1.0E-14 /y is not captured. This is a limitation of this simplified approach.

25.12 Fuel Route PSA

25.12.1 Scope of Fuel Route PSA

In the UK ABWR GDA, the fuel route is defined as following four major operation processes, namely:

- New fuel handling,
- Irradiated fuel handling,
- Irradiated fuel storage in the SFP, and
- Loading of spent fuel into the long term storage casks and subsequent transfer operations within the reactor building.

25.12.1.1 New Fuel Handling

New fuel handling is not in the scope of GDA PSA, for the following reasons:

Firstly, new fuel bundles* [in inner containers] are transported onto the ground floor of the reactor building by a small truck, the containers are then hoisted by the Reactor Building Overhead Crane (RBC) and temporarily put onto the operating floor (refuelling floor). Inner containers are then moved to the Container Stand by the RBC. New fuel bundles are moved to the New Fuel Inspection Stand (NFIS) by the RBC one by one and channel boxes are attached to the new fuel bundles. The new fuel bundles with channel boxes attached are then moved to the Fuel Preparation Machine (FPM) by the RBC.

*Fuel bundle means a fuel assembly without channel box.

Then, after being loaded into the FPM, the new fuel assemblies are moved to the spent fuel storage rack one by one by the Fuel Handling Machine (FHM). Finally, the FHM transports the fuel assemblies between the reactor core and spent fuel storage rack in SFP.

Initiating events related to operations for new fuel handling were reviewed based on the initiating event identification assessment. The identified potential initiating events are shown below:

(1) Reactivity insertion,

(2) Over-raise, and

(3) Fuel drop/collision.

For (1), criticality of new fuel is not a concern as there is no new fuel vault in the UK ABWR. Although a new fuel container may retain water, there is no way to inject water into the container. As a result, the moderator effect in the container is not considered.

For (2), as the new fuel is not activated, there is no need for fuel cooling nor concerns about radiation protection should over-raise occur.

For (3), physical damage of new fuel is not a concern, as new fuel is not activated there is no potential for offsite release and so it is not addressed further.

Therefore, risks in above steps (i.e. due to new fuel damage) are judged to be negligible.

25.12.1.2 Irradiated Fuel Handling

The irradiated fuel handling is presented in the Fuel Route PSA in this section of the report.

25.12.1.3 Irradiated Fuel Storage in the SFP

Irradiated fuel storage in the SFP is addressed by the SFP Level 1, Level 2 and Level 3 PSA (Section 25.9 of this report) for internal events and the seismic SFP PSA Level 1, Level 2 and Level 3 PSA (Section 25.11 of this report).

25.12.1.4 Spent Fuel Export

Loading of spent fuel into the long term storage casks and subsequent transfer operations within the reactor building is presented in the Fuel Route PSA in this section of the report.

Risks associated with the extended period of Spent Fuel Interim Storage (SFIS) are not considered as a part of the GDA. These will be discussed in post GDA phase.

25.12.1.5 Irradiated Fuel in the Reactor Core

Faults involving irradiated fuel in the reactor core are not considered as part of the Fuel Route PSA and are addressed by the following PSAs:

- IEAP Level 1, Level 2 and Level 3 PSA (Section 25.4, 5, 6 and 7),
- Shutdown Level 1, Level 2 and Level 3 PSA (Section 25.8),
- Internal Hazard Level 1, Level 2 and Level 3 PSA (Section 25.10), and
- External Hazard Level 1, Level 2 and Level 3 PSA (Section 25.11).

25.12.2 Initiating Events

Mechanical failure of fuel due to drop of heavy equipment is addressed in this Fuel Route PSA. Initiating events for mechanical failure of fuel are identified below and the initiating event frequencies are analysed.

25.12.2.1 Identification of Initiating Events for Dropped Load or SFIS related Operation

The initiating events for the Fuel Route PSA were identified by considering the effects of dropping all lifted items (including the spent fuel casks) based on the lifting schedules for the reactor building crane and fuel handling machine as well as the consideration of faults for spent fuel interim storage.

Furthermore, any additional cask drop scenarios which are considered in the SFP PSA (Section 25.9) are also reviewed.

(1) Initiating Event Frequency

In this subsection, the approach to quantify initiating event frequency is discussed for the identified initiating events.

Fuel drop

This initiating event can occur during refuelling operations. Therefore, this initiating event is considered in POS B-1 and POS B-2. The drop probability, fuel movement number and POS considerations are used to develop the initiating event fault tree.

Initiating event frequencies are derived by multiplying the number of fuel or other heavy load movements per year and the drop probability per movement. In the initiating event fault tree, these factors are separately modelled as basic events for each scenario/sub scenario to derive importance and to conduct sensitivity analyses for each factor. The number of fuel movements is equally divided into two and is used for the number of each POS because fuel drops can occur during POS B-1 and B-2. The initiating event fault trees are linked to each initiating event heading.

Cask drop

There are three scenarios for a cask drop:

- Drop onto cask pit or preparation pit,
- Drop onto ground floor of truck bay shaft, and
- Drop onto spent fuel storage rack in the SFP.

These initiating events can occur during cask handling operation and these are performed during at power conditions. Therefore, this initiating event is considered in POS F.

The event sequence of the cask drop onto the cask pit or preparation pit and cask drop onto the spent fuel storage rack is the same as that of fuel drop.

Initiating event frequencies are derived in a similar way to Fuel Drop using the number of cask movements per year and the drop probability per movement.

LOOP during cask lifting operation

The same LOOP frequency for IEAP is used for this initiating event. The cask handling operation is performed during at Power conditions. Therefore, this initiating event is considered in POS F.

25.12.2.2 Other Scenarios

In this section, other related events and secondary effects are discussed:

- Cask or other heavy load drops down onto the equipment shaft,
- ECCS compartment,
- RCW piping,
- Containment structure, and
- Inside drywell.

(1) Cask or other heavy load drop down the equipment shaft

The following systems have their piping in the equipment shaft (large component entrance).

- D22: Airborne Activity Monitoring System
- P11: Make-up Water Purified System (MUWP)

Even if these two systems are damaged by the dropped load, an initiating event does not occur. In the truck bay shaft, there are no SSCs (piping or cables) for safety related systems. Therefore, any secondary effects due to heavy load drop down the equipment shaft are not a safety related concern.

However, MUWP piping on the ground floor is potentially damaged by a heavy load drop in the truck bay. If MUWP piping is damaged by the drop, dropped load induced flooding is possible. The flood source is 2,800 m^3 from the Purified Water Storage Tank. In this case, the flooded water goes into a corridor of the reactor building bottom floor. This flooding volume is bounded by that of the suppression pool flooding case and it does not result in a water level above 4 m. Therefore, the flooding causes no initiating event and no disabling of any safety measures.

(2) ECCS compartment

The heaviest dropped load (i.e. the cask) on the truck bay shaft does not penetrate the underground floor. This is confirmed by Finite Element Method (FEM) analysis.

The lifting rails do not cross the safety divisions.

There is no important piping or electrical cable/trays in the truck bay shaft etc.

Therefore, there is no disabling of any safety measures.

(3) RCW piping

It is evaluated that there is no credible initiating event due to a drop onto RCW piping as the rails do not cross the safety division and lifted equipment is already out of service due to outage maintenance.

(4) Containment structure

Heavy equipment is not lifted directly across the PCV in the plant at Power condition as the RCCV head is covered by reactor well shield plug in all at Power conditions.

In shutdown conditions, it is assumed the PCV boundary is already opened therefore containment structure integrity to prevent leakage is not a concern.

(5) Inside drywell

According to the internal hazard assessment, the following lifting items in the drywell have the potential to cause a LOCA potential:

- SRVs,
- MSIVs (inboard), and
- DWC (Drywell Cooler) fans.

Lifting operations are only carried out during outage operations (POS B).

These items are lifted along the monorail and unloaded on the slope and then moved out to outside of the PCV through the equipment hatch. Under the monorail for SRVs, MS (Main Steam) piping is located. In the case of a drop on the MS line, there is no effect even if the MS line is damaged because MS line plug isolates the line. Therefore, potential drop locations which may cause a LOCA are limited. The identified potential drop locations contain the following systems which may cause a LOCA.

- Feedwater line
- RHR line
- HPCF
- Other small piping

The potential drop location which causes consequential LOCA is conservatively assumed to be about 10 percent of total monorail length and the total probability of LOCA due to dropped load inside drywell in

POS B is derived. The FDF due to a dropped load induced LOCA inside the drywell is sufficiently low to be ignored as a risk contributor of the plant.

25.12.2.3 Other Aspects

(1) Re-criticality in the cask/canister

The following aspects for dry cask storage risk are mentioned in the NUREG-1864 [Ref-25.118] concerning areas of uncertainty relevant to the progression of severe accidents.

> *Several calculations were performed to evaluate MPC performance. The first observation is that the compression/expansion study demonstrates that there are no credible compression/expansion scenarios which lead to an increase in the eigenvalue to anywhere near the 5 % sub-critical margin. Similar analyses are shown in Reference 48. The second observation is that there are no low density, optimal moderation configurations which means that pool draining events will not lead to a reduction in the subcritical margin. Figure 19 and the work in Reference 49 confirm this by showing the predicted eigenvalue at the upper bound 95th percentile confidence interval with reduced water density. As is evident from the figure, even when changes in water density are considered, the fuel in the MPC is highly sub-critical. In summary, the criticality analyses performed for this study, while limited in scope, show similar trends as other studies by confirming that the configuration is under-moderated and highly subcritical.*
>
> *When the MPC is dry, criticality is not physically possible under any conditions in the absence of other neutron sources. This is true because the original fuel enrichment was limited to 5 % weight, which is a limit based in part on the requirement that unmoderated criticality is not possible, even for BWR fuel at its most reactive point in life. This position is based on experimental data and calculations in Reference 50.*
>
> *The borated aluminium plate used in the MPC-68 basket is an effective means of criticality control; criticality is not allowed whether moderated or unmoderated. Currently several types of the basket designs are available such as borated aluminium plates are attached on the basket plates or the borated aluminium plate itself forms basket structure. The attached borated aluminium is assumed here for discussion (detailed design of the basket will be developed during site specific stage). No credible scenario which would eliminate the borated aluminium plating and leave the geometry of the fuel intact has been identified. Even if the borated aluminium plate were to separate from the basket structure, they would have limited room to move and, once the canister is sealed, there is no credible means by which they could fall out of the basket. If the contents of the MPC-68 were to somehow relocate, the borated aluminium plated plates would relocate with the debris. Likely, this postulated configuration would be highly subcritical because of the boral and because a bed of debris is not an optimal geometry. At the enrichments used in BWR fuel, a square pitch lattice is the optimal geometry for criticality. In the absence of moderation, experiments and calculations have demonstrated that criticality is not possible at the enrichments currently used in light-water reactor (LWR) fuel.*

The cask/canister system to be used in UK ABWR is not fixed but it is available to select the same design as that of NUREG-1864 or similar design with the same safety functions. Therefore, re-criticality is not concern for the above reasons.

(2) Cask Mis-loading

When the operator mis-loads the relatively high decay heat level spent fuel into the cask, the potential for the fuel to heat-up and fail should be considered.

Any temperature increase would be detected by a thermocouple and it initiates an alarm.

If the operator can recognise the mis-loading, there is potential for it to be recovered within the time margin. Even when the mis-load is not identified, it is considered that the internal pressure of the canister remains under the allowable design pressure of the canister (mis-load of 8 fuel assemblies of 1 year cooling is assumed in this evaluation). Therefore, this event is not explicitly quantified.

25.12.3 Event Sequence Analysis

Potential mechanisms for fission product release from the damaged spent fuel, due to dropped load accident conditions, are identified to conduct the fuel route/dropped load PSA. For these events, the containment boundary does not prevent fission product releases from SFP since the spent fuel damage due to dropped load occurs outside of containment.

In this section, each fission product release path due to reactor ventilation conditions is described.

25.12.3.1 Event Sequence Analysis for Fuel Drop

The event sequences considered for the fuel drop accident are outlined in Figure 25.12.3-1. The boxes with a red outline show the release path for each event sequence.

(1) Event sequence and credited mitigation features to prevent the fuel damage

In the fuel drop event, it is assumed spent fuel will be damaged by the dropped load.

No SSCs are credited in this accident sequence before fuel damage.

(2) Accident progression and credited mitigation features to reduce offsite releases

Fission products are released from the damaged spent fuel. The secondary containment function can prevent direct release of the Fission Products by isolation of R/A (Reactor Area) HVAC and initiation of SGTS filtering.

Upon receiving an emergency signal (“R/A exhaust radioactivity high” or “refuelling area radioactivity high”), the R/A HVAC is automatically stopped and the R/A isolation dampers automatically close, thereby isolating the R/A and preventing exfiltration of the radioactive gas to the environment. The SGTS also automatically starts thereby maintaining a reduced pressure in the secondary containment to control the discharge of radioactive substances and process the gaseous effluent through the filter train to remove any airborne iodine and particulates. The release to the environment occurs via the stack.

Three further event sequences are considered:

- R/B is isolated but SGTS initiation fails. The release to the environment occurs via natural leakage through the secondary containment boundary. This release is unfiltered and at ground level.
- R/B is not isolated but SGTS initiation occurs. When the R/A HVAC are not isolated, the effectiveness of the SGTS is limited and the consequences of this event sequence are represented by the following event sequence.
- R/B is not isolated and SGTS initiation fails. The release to the environment occurs via the R/A HVAC and is assumed to be entrained into the building wake.

The above described event sequence and accident progression is like that of two other events, except for the POS and the amount of damaged fuel:

- Cask drop onto spent fuel storage rack in the SFP and

- Cask drop onto cask pit or preparation pit

Therefore, the same event tree structure is used to assess these two cask drop scenarios.

25.12.3.2 Event Sequence Analysis for Cask Drop onto Truck Bay Shaft

The event sequence of the cask drop onto the truck bay shaft is outlined in Figure 25.12.3-2. The boxes with a red outline show the release path for each event sequence.

(1) Event sequence and credited mitigation features to prevent the fuel damage

Potential mechanisms for fission product release from the damaged spent fuel in the cask due to cask drop accident in the truck bay shaft are almost the same as for Fuel Drop except for the effect of the impact limiter.

An impact limiter is installed in the ground floor of the truck bay to mitigate the dropped load impact and to prevent canister containment failure. The impact limiter and canister are credited in this event.

(2) Accident progression and credited mitigation features to prevent offsite release

The impact limiter can prevent failure of the cask in the dropped load event. In this event sequence, there is no release of fission products.

If the impact limiter fails to prevent damage to the fuel and cask, the accident progression and mitigation claims are the same as the fuel drop accident scenario. R/A HVAC and SGTS functions are credited in these accident progression sequences in the same way.

25.12.3.3 Event Sequence Analysis for LOOP in Cask Handling

The event sequence of a LOOP during cask handling is outlined in Figure 25.12.3-3. The boxes with a red outline show the release path for each event sequence.

(1) Event sequence and credited mitigation features to prevent the fuel damage

If a LOOP occurs during cask handling between the cask pit and the preparation pit, an EDG (one safety division) provides AC power to the RBC. If the RBC mechanically fails or the EDG fails, the canister cooling system and backup cooling system, supported by the BBG, are connected to the cask to maintain the cask cooling. If both these systems fail, spent fuel in the cask is assumed to heat up and undergo clad failure resulting in release of all the activity in the gap and plenum regions of the fuel rods. In this case, radioactive gases and airborne activity associated with evaporated water are assumed to leak from the cask.

An EDG for one safety division is credited in the event tree for this initiating event. After that, operator action to connect the canister cooling system is credited.

The backup canister cooling system is not credited as these are dependent on the operator action. Operator action to move the RBC manually back to the cask pit or preparation pit is under consideration. This operator action is not credited because a detailed procedure is not defined yet.

(2) Accident progression and credited mitigation features to prevent offsite release

Where the cask can be returned to the cask pit or preparation pit, there is no release of fission products.

If the cask cannot be moved but the cask can be cooled by connection of the cask cooling system, there is no release of fission products.

Where neither system is successful, it is assumed the fuel in the cask fails. The subsequent event sequence progression is the same as for the fuel drop accident scenario. R/A HVAC and SGTS functions are credited in the event sequences.

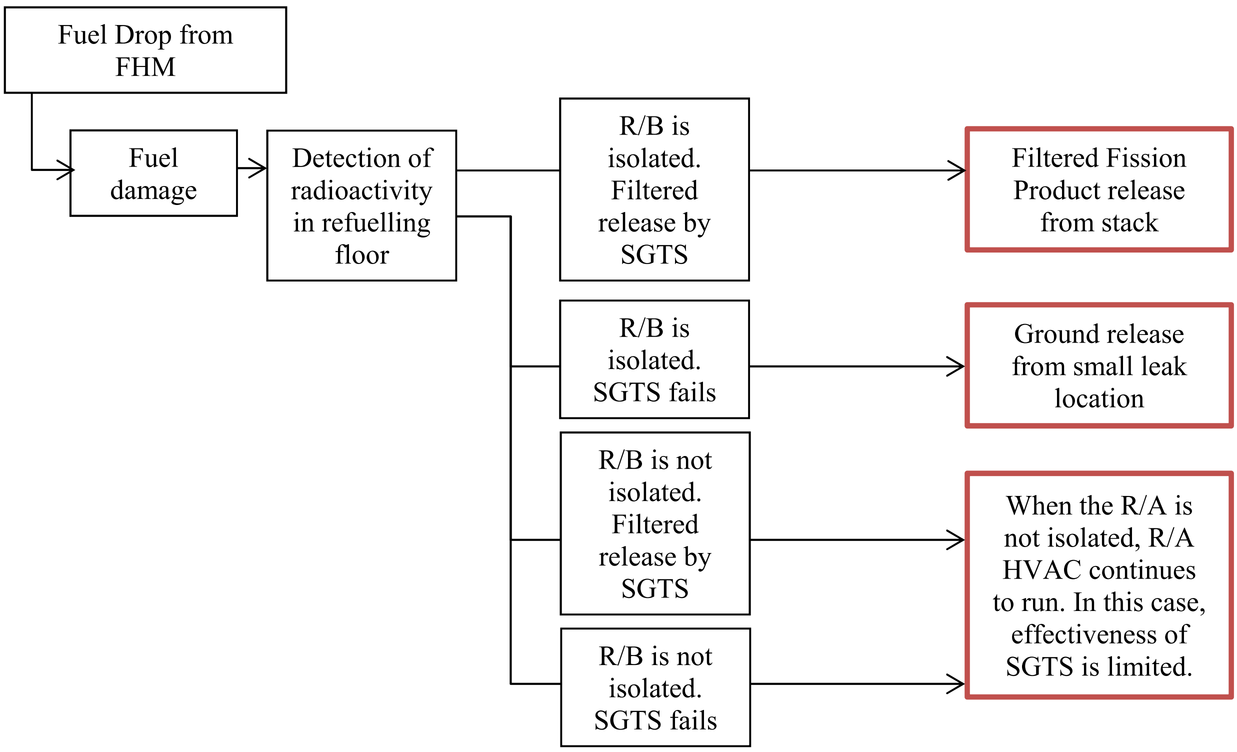


Figure 25.12.3-1 Outline of Accident Progression for Fuel Drop

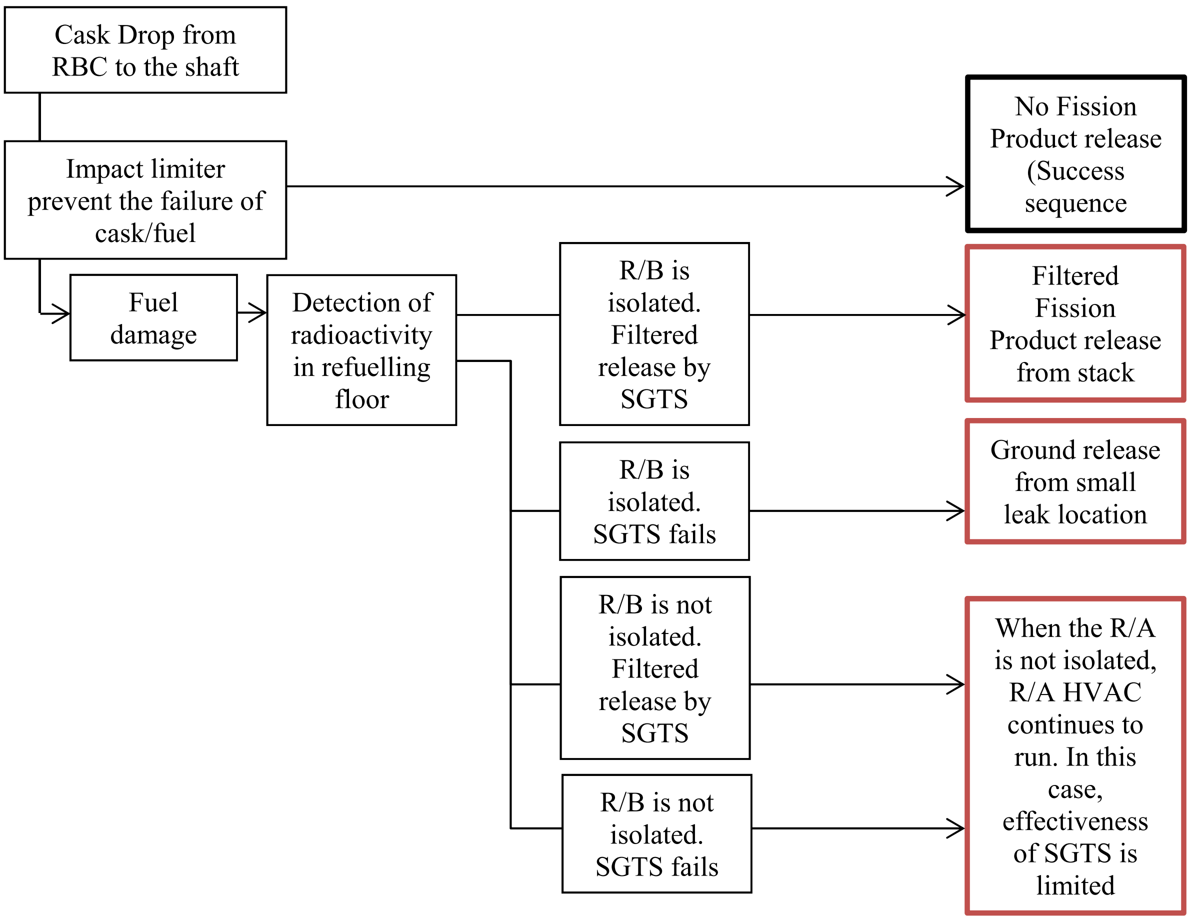


Figure 25.12.3-2 Outline of Accident Progression for Cask Drop from RBC to the Shaft

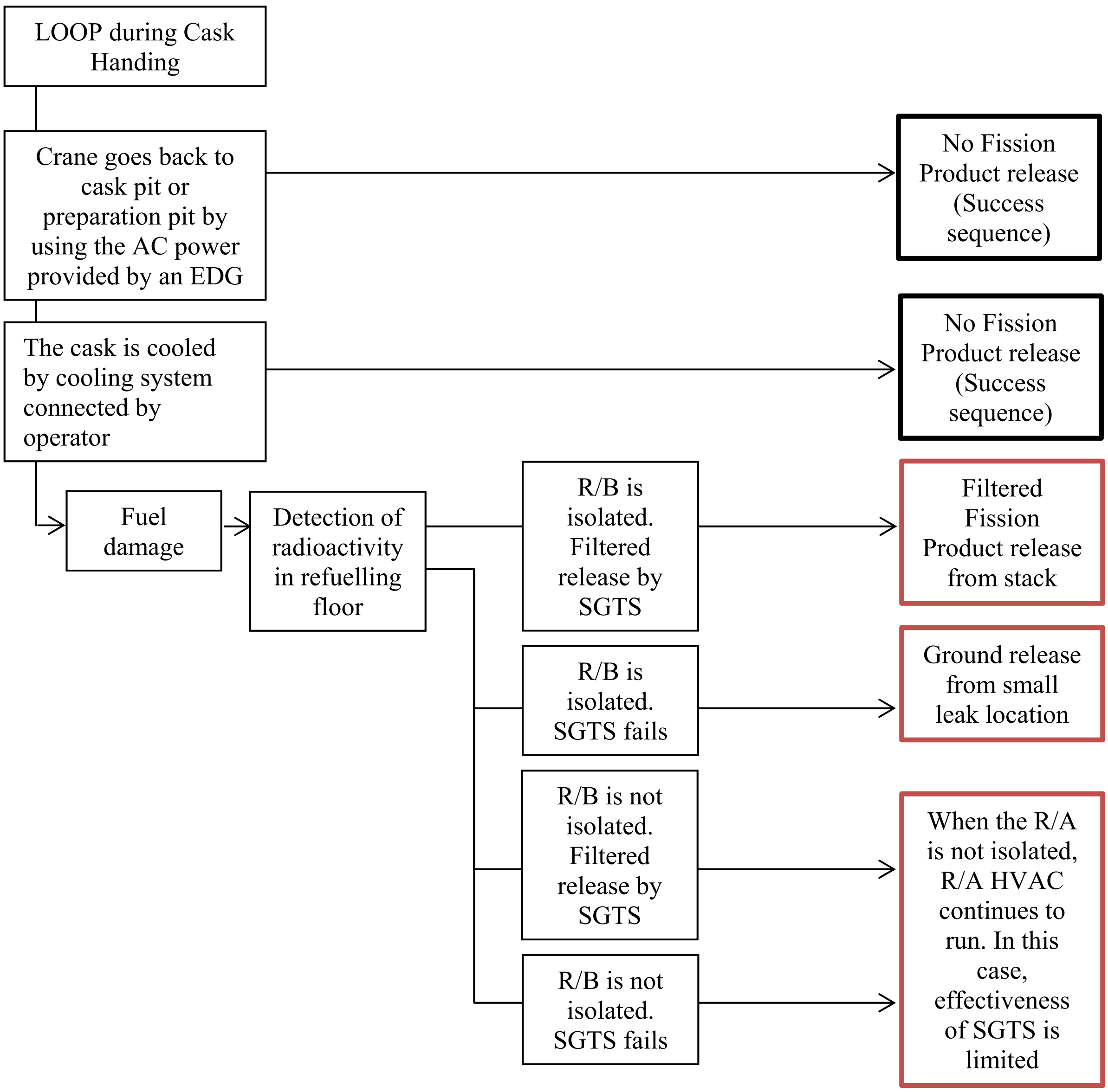


Figure 25.12.3-3 Outline of Accident Progression for LOOP in Cask Handling

25.12.4 Results (Fuel Route Level 1 PSA)

The purpose of this section is to document the calculation of the RF (release frequency) due to events in all dropped load fuel damage sequences.

No sequences leading to fuel melt have been identified in the event sequence analysis and all faults occur outside the containment boundary. As a result, there is no requirement for a Level 2 PSA and the release frequency is taken directly into the Level 3 PSA.

25.12.4.1 Calculation of Release Frequency

A total of 18 event progression sequences are quantified as release end states of the event trees. Figure 25.12.4-1 illustrates the RF contribution for all accident sequences in the form of a bar chart.

All accident sequences which contribute to RF are described below.

(1) Fuel Damage (FD) scenario

(i) FD01: an accident progression sequence due to fuel drop

- Local fuel damage due to dropped load.
- R/A HVAC successfully isolated.
- Fission products are filtered by SGTS and released to the environment from the stack.

(ii) FD02: an accident progression sequence due to fuel drop

- Local fuel damage due to dropped load.
- R/A HVAC successfully isolated.
- SGTS fails to filter the fission products and they are released to the environment.

(iii) FD03: an accident progression sequence due to fuel drop

- Local fuel damage due to dropped load.
- R/A HVAC fails to isolate.
- SGTS fails to filter the fission products and they are released to environment.

(2) Cask Drop onto spent fuel storage rack (CD-SFP) scenario

(i) CD-SFP01: an accident progression sequence due to cask drop onto spent fuel storage rack.

- Local fuel damage due to cask drop on to spent fuel storage rack.
- R/A HVAC successfully isolated.
- Fission products are filtered by SGTS and released to environment from the stack.

(ii) CD-SFP02: an accident progression sequence due to cask drop onto spent fuel storage rack

- Local fuel damage due to cask drop on to spent fuel storage rack.
- R/A HVAC successfully isolated.
- SGTS fails to filter the fission products and they are released to environment.

(iii) CD-SFP03: an accident progression sequence due to cask drop onto spent fuel storage rack

- Local fuel damage due to cask drop on to spent fuel storage rack.
- R/A HVAC fails to isolate.
- SGTS fails to filter the fission products and they are released to environment.

(3) Cask Drop onto cask pit or preparation out (CD-CP) scenario

(i) CD-CP01: an accident progression sequence due to cask drop onto cask pit or preparation pit

- Local fuel damage due to cask drop on to cask pit or preparation pit.
- R/A HVAC successfully isolated.
- Fission products are filtered by SGTS and released to environment from the stack.

(ii) CD-CP02: an accident progression sequence due to cask drop onto cask pit or preparation pit

- Local fuel damage due to cask drop on to cask pit or preparation pit.
- R/A HVAC successfully isolated.
- SGTS fails to filter the fission products and they are released to environment.

(iii) CD-CP03: an accident progression sequence due to cask drop onto cask pit or preparation pit

- Local fuel damage due to cask drop on to cask pit or preparation pit.
- R/A HVAC fails to isolate.
- SGTS fails to filter the fission products and they are released to environment.

(4) Cask Drop onto truck bay shaft (CD) scenario

(i) CD02: an accident progression sequence due to cask drop onto truck bay shaft

- Cask drops onto truck bay shaft
- Impact limiter successfully decelerates the impact.
- Canister fails to maintain the boundary.
- R/A HVAC successfully isolated.
- Fission products are filtered by SGTS and released to environment from the stack.

(ii) CD03: an accident progression sequence due to cask drop onto truck bay shaft

- Cask drops onto truck bay shaft
- Impact limiter successfully decelerates the impact.
- Canister fails to maintain the boundary.
- R/A HVAC successfully isolated.
- SGTS fails to filter the fission products and they are released to environment.

(iii) CD04: an accident progression sequence due to cask drop onto truck bay shaft

- Cask drops onto truck bay shaft
- Impact limiter successfully decelerates the impact.
- Canister fails to maintain the boundary.
- R/A HVAC fails to isolate.
- SGTS fails to filter the fission products and they are released to environment.

(iv) CD05: an accident progression sequence due to cask drop onto truck bay shaft

- Cask drops onto truck bay shaft
- Impact limiter fails to decelerate the impact and the fuel assemblies are damaged.
- R/A HVAC successfully isolated.
- Fission products are filtered by SGTS and released to environment from the stack.

(v) CD06: an accident progression sequence due to cask drop onto truck bay shaft

- Cask drops onto truck bay shaft
- Impact limiter fails to decelerate the impact and the fuel assemblies are damaged.
- R/A HVAC successfully isolated.
- SGTS fails to filter the fission products and they are released to environment.

(vi) CD07: an accident progression sequence due to cask drop onto truck bay shaft

- Cask drops onto truck bay shaft
- Impact limiter fails to decelerate the impact and the fuel assemblies are damaged.
- R/A HVAC fails to isolate.
- SGTS fails to filter the fission products and released to environment.

(5) LOOP during cask handling operation (LOOP-DR) scenario

(i) LOOP-DR03: an accident progression sequence due to LOOP during cask handling operation

- LOOP occurs during cask handling operation
- EDG fails to deliver the AC power to RBC.
- CCS fails to cool the spent fuel. The spent fuel is heated up and damaged.
- R/A HVAC successfully isolated.
- Fission products are filtered by SGTS and released to environment from the stack.

(ii) LOOP-DR04: an accident progression sequence due to LOOP during cask handling operation

- LOOP occurs during cask handling operation
- EDG fails to deliver the AC power to RBC.
- CCS fails to cool the spent fuel. The spent fuel is heated up and damaged.
- R/A HVAC successfully isolated.
- SGTS fails to filter the fission products and they are released to environment.

(iii) LOOP-DR05: an accident progression sequence due to LOOP during cask handling operation

- LOOP occurs during cask handling operation
- EDG fails to deliver the AC power to RBC.
- CCS fails to cool the spent fuel. The spent fuel is heated up and damaged.
- R/A HVAC fails to isolate.
- SGTS fails to filter the fission products and they are released to environment.

25.12.4.2 Interface with Level 3 PSA

The 18 event progression sequences that are quantified in the event trees are condensed into 8 sequence groups, based on the similarity of releases to the environment. Table 25.12.4-1 illustrates how the sequences are combined.

Table 25.12.4-1 Sequence Group for each Event Progression Sequence

No.	Event progression sequence	Sequence group
1	FD01	FS: Fuel Handling Accident with SGTS successful
2	FD02	FH : Fuel Handling Accident with SGTS unsuccessful
3	FD03	
4	CD-SFP01	HLPS: Heavy Load (e.g. cask) drop into spent fuel pool with SGTS successful
5	CD-SFP02	HLPF: Heavy Load (e.g. cask) drop into spent fuel pool with SGTS unsuccessful
6	CD-SFP03	
7	CD-CP01	CDPS: Cask drop onto cask pit or preparation pit with SGTS successful
8	CD-CP02	CDPF: Cask drop onto cask pit or preparation pit with SGTS unsuccessful
9	CD-CP03	
10	CD02	CDTS: Cask drop onto truck bay shaft or LOOP with SGTS successful
11	CD03	CDTF: Cask drop onto truck bay shaft or LOOP with SGTS unsuccessful
12	CD04	
13	CD05	CDTS: Cask drop onto truck bay shaft or LOOP with SGTS successful
14	CD06	CDTF: Cask drop onto truck bay shaft or LOOP with SGTS unsuccessful
15	CD07	
16	LOOP-DR03	CDTS: Cask drop onto truck bay shaft or LOOP with SGTS successful
17	LOOP-DR04	CDTF: Cask drop onto truck bay shaft or LOOP with SGTS unsuccessful
18	LOOP-DR05	

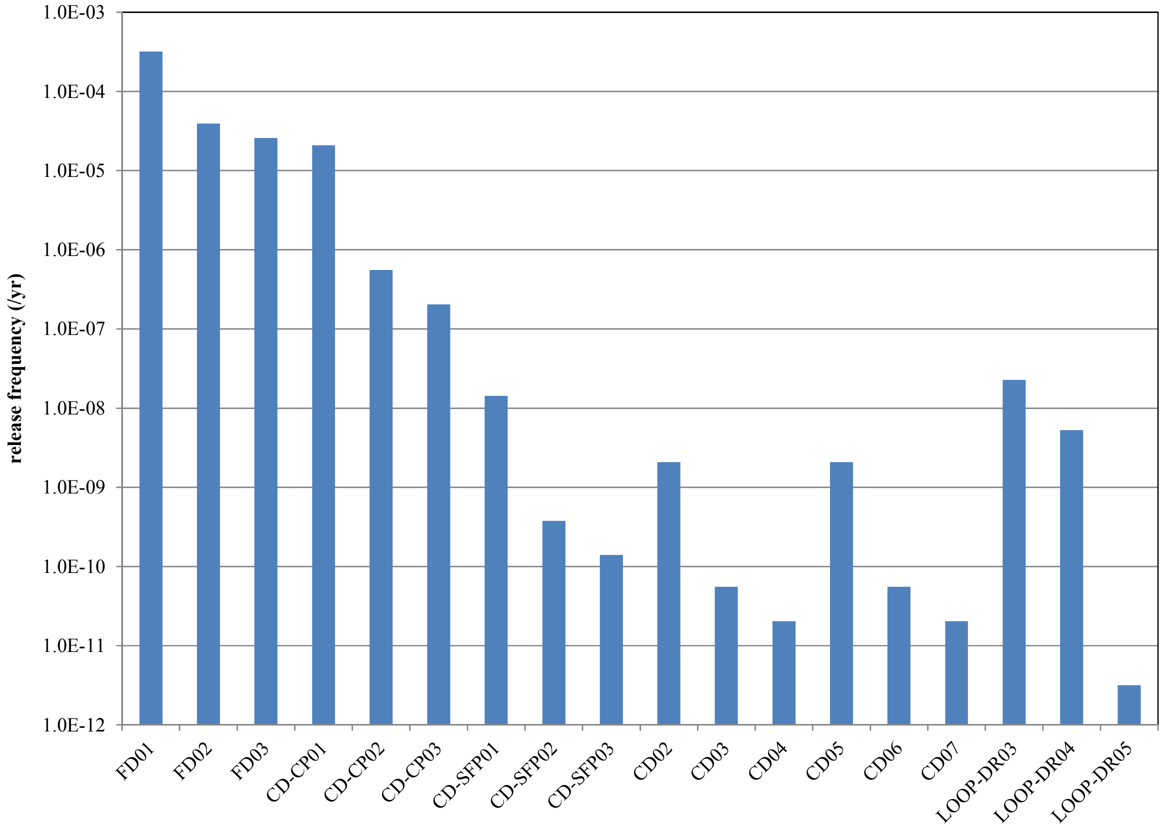


Figure 25.12.4-1 Contribution to RF by all Accident Progression Sequence

25.12.5 Analysis for Level 3 PSA

The representative sequence for each sequence group defined in sub-section 25.12.4.2 is considered in the Design Basis Assessment (DBA). In the DBA, a deterministic dose evaluation is reported [Ref -117]. This dose is used in the assessment against facility dose bands in the Level 3 PSA, to avoid duplication of source term and dose calculations for the same faults. It is noted that both the source term calculations and the deterministic dose calculations are performed on a conservative basis, as required by the DBA.

25.12.5.1 Representative Source Term

In this section, the representative source term for each sequence group is briefly described. The following description is the same as in the above-noted topic report. In the document, evaluations for the mitigated and unmitigated cases are performed, i.e. cases with R/A isolation and successful operation of the SGTS and cases with no R/A isolation.

These evaluations are utilised as the source term for fuel drop scenario in this Fuel Route/Dropped Load PSA.

(1) Accident progression and credited mitigation features to prevent offsite release

The sequence of fuel drop is described as followings.

(1) During a refuelling operation, fuel drop is assumed to occur when fuel assemblies are being manipulated over the reactor core for refuelling, and to fall on top of the core impacting a group of four assemblies.

(2) Some rods in both the dropped and struck assemblies fail, releasing radioactive gases into the reactor water.

(3) Gases pass from the water to the surface of water in reactor well area.

(4) For the mitigated case, the R/B Operating deck high radiation alarm alerts plant personnel and initiates isolation of the R/A HVAC and the start of the SGTS is automatic. The unmitigated case does not credit these alarms, the R/A HVAC isolation or the SGTS operation.

(2) Mechanical Damage Fraction

Fuel Drop

The amount of fuel damaged because of a drop of an irradiated fuel assembly is evaluated and it is determined that a maximum of 1.6 fuel assemblies are damaged. Since there are 872 fuel assemblies in the UK ABWR core the total core damage fraction is equivalent to 1.83E-03 (equal to 1.6 fuel assemblies/872 fuel assemblies).

Cask Drop

(a) Cask drop onto SFP

The number of damaged fuel assemblies due to cask drop onto spent fuel storage rack is conservatively calculated based on the J-ABWR dimensions.

Case 1: Vertical drop case

The cask is assumed to drops without toppling on the spent fuel storage rack. In this case, the cross section of the cask bottom (area of φ2.6 m) may impact the spent fuel assemblies. This cross section covers about 220 fuel assemblies in the rack. This 220 is assumed to the number of damaged fuel assemblies due to a cask drop without toppling.

Case 2: Cask toppling case

The cask is assumed to drops with toppling on the spent fuel storage rack. In this case, the cask wall of the cask side (about 5.5 m high) may impact the spent fuels assemblies. The potential impact area is assumed to be corresponding to the area equal to 5 fuel assemblies wide multiplied by a length of 5.5 m. This cross section covers about 185 fuel assemblies in the rack. This 185 is assumed to the number of damaged fuel assemblies due to a cask drop with toppling.

Therefore, the 220 is conservatively assumed to be the number of damaged fuel assemblies due to cask drop onto spent fuel storage rack.

As mentioned in the case of fuel drop, the amount of fuel damaged because of a drop of “an irradiated fuel assembly” is evaluated and it is determined that a maximum of 1.6 fuel assemblies are damaged.

Based on the above, the release of the fuel drop sequence multiplied by the ratio (220/1.6=137.5) is used as the source term for cask drop onto spent fuel storage rack.

(b) Cask drop onto cask pit or preparation pit

The number of damaged fuel assemblies due to cask drop onto cask pit or preparation pit is conservatively calculated based on the design specification of the cask/canister.

Although the impact limiter is installed in the bottom of both cask pit and preparation pit, canister confinement function cannot be credited before the welding of the canister lid. The function of the impact limiter is mitigation of the impact to protect the canister confinement function. Therefore, the impact limiter is not credited to maintain the integrity of fuel assemblies. Therefore, the all spent fuel assemblies installed in the cask/canister are conservatively assumed to be damaged due to the drop's impact.

As the basic specification, the number of installed fuel assemblies in the cask/canister is 89. The release of the fuel drop sequence multiplied by the ratio (89/1.6=55.6) is used as the source term for cask drop onto the cask pit or preparation pit.

(c) Cask drop onto truck bay shaft

In NUREG-1864 [Ref-25.118], source term was determined using the following equation:

$F_{rel} = F_{rod} \times F_{RC} \times F_{CE}$

Where,

F_{rel}: The fractional release from the cask

F_{rod}: the fraction of rods contributing to the release.

F_{RC}: the fraction of material released from a single rod to the cask environment.

F_{CE}: the fraction of respirable radioactive material in the cask atmosphere that is released to the environment and F_{relk} = fraction of the radioactive material in the cask that is released to the environment in the respirable size range.

Although NUREG-1864 [Ref-25.118] concluded that the values described in Table 25.12.5-1 should be used for the consequence analysis, it has large range of parameter uncertainty and the model has not been benchmarked against experimental data.

Table 25.12.5-1 Values of Release Fractions [Ref-25.118]

Table D.7. Values of Release Fractions F_{rods}, $F_{RC,k}$, $F_{CE,k}$, and $F_{rel,k}$*

(10-year-old cooled 10x10 Atrium fuel, 30.5 meter (100-foot) drop of a HI-STORM 100 cask, all of the rods fail with an average of 5 breach locations per rod)

Chemical Element Group (k)	Inventory	F_{rods}	$F_{RC,k}$	$F_{CE,k}$	$F_{rel,k} = F_{rods}F_{RC,k}F_{CE,k}$ respirable
Noble Gases (k=1)	ORIGEN	1	0.12	1	0.12
Particles (k=2)	ORIGEN	1	7×10^{-5} to 1.2×10^{-2}	0.1	7×10^{-6} to 1.2×10^{-3}
CRUD (k=3)	0.72Ci/rod	1	0.015	0.1	0.0015

In the above parameter, $F_{RC,\,2}$ has relatively large range (from 7×10^{-5} to 1.2×10^{-2}). This is explained as "The most conservative position is to use the highest release fraction from Table D.6 ($F_{RC,\,k}$), but that may be considerably higher than a realistic value [Ref -118]". In addition, the drop height is about 10 m lower than the case of NUREG-1864 [Ref-25.118].

Therefore, $F_{RC,\,2}$ is assumed in order to estimate a realistic source term of the cask drop for this fuel route/dropped load PSA. Other parameters are set to the same value as shown in the Table 25.12.5-1.

LOOP during cask lifting operation

Source term analysis for a LOOP during cask handling is also based on the same equation as described in this section. However, the following conditions are significantly different for a LOOP compared to from a cask drop.

(1) The number of fuel assemblies which are damaged due to heating up is limited because decay heat level of certain fuel assemblies is sufficiently small.

(2) Breach of a heated-up fuel rod is supposed to be shorter and narrower than that of cask drop case because these are no drop or collision impact.

(3) Integrity of the cask and canister is maintained because of no drop or collision impact. Therefore, the release path from the canister is only the gap between the canister and unsealed canister lid which located at the upper side of the canister.

(4) The postulated driving force for the particle Fission Products release is convection flow and thermal expansion of gas inside the canister. This driving force is supposed to be significantly smaller than that of a cask drop because there is no drop or collision impact. In addition, the convection flow and thermal expansion of gas should defy gravity.

Therefore, each parameter is changed as follows:

- F_{rod} for all chemical element groups is assumed to be 0.5 because of the above (2).
- $F_{RC,2}$ is assumed to be 1.2×10^{-4} and $F_{RC,3}$ is assumed to be 1.5×10^{-3} because of the above (1).
- $F_{CE,2}$ is assumed to be 0.01 because of the above (3) and (4).

(3) Release from Damaged Fuel

All the activity in the gap and plenum regions of the failed rods is assumed to release instantaneously at the location where the damage occurs (i.e. either into the SFP, a water filled cask or an air-filled cask).

It is assumed that all the gap activity in the damaged rods is released at the time of the accident. This is conservatively assumed to consist of:

- 4 percent of the total noble gases other than Kr-85, and 35 percent of the Kr-85,
- 8 percent of the I-131, 23 percent of the I-132, and 5 percent of other Halogens, and
- 46 percent of the other radionuclides in the gap inventory as Alkali metals.

These assumptions are based on the DRAFT REGULATORY GUIDE DG-1199 [Ref-25.119] and TECHNICAL BASIS FOR REVISED REGULATORY GUIDE 1.183 FISSION PRODUCT FUEL-TO-CLADDING GAP INVENTORY [Ref-25.120]. This fraction is calculated by the FRAPCON code considering the Halden reactor data, NEDO-33163 [Ref-25.121]. Also, the values given are conservatively based on the maximum value for both PWR and BWR fuel. Hitachi-GE also compared the release fractions between the LWR fuel, GE14 and GE8x8 fuel in other reports. Based on these data, Hitachi-GE decided on the most conservative values based on the 3 types of fuel.

The gap inventory of Alkali metals assumes Cs-134 is used to represent the alkali metals. The Cs-134 is released to the gap in the fuel rod at power operation. Because the boiling point of Cs-134 is around 700 °C, Cs-134 is selected as the conservative case for all alkali metals. Therefore, Hitachi-GE conservatively assumed that the alkali metals are released from the gap even if the plant is in an outage condition (i.e., the temperature of the fuel is same as the outage water temperature.).

As the fuel fragments generated by impact are much larger than an aerosol, they do not disperse onto the pool surface; therefore, nuclides other than noble gases, halogens and alkali metals are excluded from the calculation for source term.

(4) Release to the environment

In all sequences, the retention effect in the building is assumed to correspond to a DF: 10 for nuclides other than noble gas and iodine.

In some sequences, the R/B ventilation isolates and the SGTS initiates from a "Reactor building operational floor high radiation signal," detecting radioactivity released onto the pool surface. Since the R/B ventilation system isolates immediately and the start of the SGTS is automatic, the contamination is processed by the SGTS and subsequently released to the environment from the plant stack. Radioactivity decay in the reactor building corresponding to the SGTS flow rate is considered. The SGTS is credited with a filter efficiency of 99.9 percent for all iodine chemical species and particulates.

No other retention mechanisms are claimed.

25.12.5.2 Conditional Consequences Assessments

The representative doses for the Fuel Route sequence groups, for assessment against the facility dose bands, are taken from the Topic Report on Non-Reactor Faults and Reactor Lower Dose Sequences [Ref-25.122]. As previously discussed, these doses are calculated using the more conservative methodology of DBA. At this time, no allowance has been made for the more realistic source term, dispersion and dose uptake characteristics appropriate for PSA assessments.

The results of the probabilistic consequences calculations, to be used in the assessment against Target 7 and Target 9, are summarised in section 25.13. These calculations have been performed using the same bounding radionuclide releases as the deterministic dose calculations but use the same methodology as the probabilistic consequences calculations for release categories resulting in fuel melt.

25.12.5.3 Results for Level 3 PSA

The results of the Level 3 PSA are presented with other non-fuel melt sequence groups/fault groups in Section 25.13.

25.12.6 Uncertainty and Sensitivity Analysis

25.12.6.1 Sensitivity Analysis Results (Individual Sensitivity Cases)

Sensitivity Analyses have been performed for each key safety function using quantitative and/or qualitative methods (see Section 25.7.5.2 (b)). Table 25.12.6-1 shows the summary of the sensitivity analyses.

Table 25.12.6-1 Sensitivity Analysis for Fuel Cooling

No.	Sensitivity Analysis Case	Summary of analysis results
1	SGTS success criteria	The difference between the base case results and sensitivity case results for some sequences are +1,000 percent or more. Although it seems significant difference, the effected case RFs for CD, CD-CP and CD-SFP scenarios are 1.0E-07 (/y) to 1.0E-11 (/y) order of magnitude. For CD-CP scenario, although the resulting frequency is relatively high, the consequence is small (1 mSv order of magnitude). This sensitivity case resulted that there is no significant impact even though these originally small effect scenarios may have slightly higher frequency.
2	Conservatism of human failure event	A sensitivity analysis is performed if the realistic HEP for backup CCS is used in the LOOP-DR scenario. RF is decreased by 32.3 percent.
3	Initiating Event Frequency related to Cask Drop	The realistic number of cask handling is calculated based on the core design information and so on. 92 percent of RFs are decreased when the realistic number of cask handling is considered. Although it seems significant difference, the base case RFs for CD scenario and CD-SFP scenario are 1.0E-08 (/y) to 1.0E-09 (/y) order of magnitude. For CD-CP scenario, although the resulting frequency is relatively high, the consequence is small (1 mSv order of magnitude). This sensitivity case resulted that these originally small effect scenarios may become less severe.
4	Decrease of conditional probability due to reactor challenge	The RFs of sensitivity cases are proportional to re-calculated initiating event frequencies. Total initiating event frequency for FD scenario (i.e. FD01 sequence) is 3.19E-04 (/y). The increase of total initiating event frequency is 5.7 percent derived from following equation. (1.81E-05 + 3.19E-04) / 3.19E-04 = 1.057 FD02 and FD03 scenarios are also increased as the same level. These increases are not significant impact on the RFs.
5	Appropriateness of the failure rate and the test interval time of digital control equipment for SGTS and R/A HVAC	The difference between the base case results and sensitivity case results are -49 percent to -95 percent. Although it seems significant difference, the base case RFs for these scenarios are 1.0E-08 (/y) to 1.0E-09 (/y) order of magnitude or significant small release. For LOOP-DR scenario, initiation signal for R/A HVAC succeeds because of fail-safe I&C design. Therefore, failure rate for digital control of R/A HVAC has no effect on the LOOP-DR05 sequence. This sensitivity case resulted that these originally small effect scenarios may become less severe.

25.12.6.2 Uncertainty Analysis

Uncertainty analysis is performed using the Monte Carlo sampling method which generates a probability density function and a cumulative probability function for Release Frequency (RF). The uncertainty distribution and error factors are captured in the type code (TC) table of the CAFTA database. For CCF, the alpha-factor method is used to quantify CCFs in the UK ABWR PSA and is preferred as it enables more accurate modelling of uncertainty in the CCF parameters.

A sample size of 100,000 is used to generate the associated results from the cutsets with a truncation value consistent with the point estimate calculation for each scenario. The mean values generated based on the sample size are as follows:

- Fuel Drop (FD) : 3.93E-04 /y
- Cask Drop onto spent fuel storage pool (CD-SFP) : 1.44E-08 /y
- Cask Drop onto cask pit or preparation pit (CD-CP) : 2.23E-05 /y
- Cask Drop (CD) : 4.51E-09 /y
- LOOP during cask handling (LOOP-DR) : 2.82E-08 /y

There is no significant difference between the above mean values and point estimation results. Therefore, the comparison between the targets and this PSA results can be discussed by point estimation.

25.12.7 Key Assumptions and Study Limitations

Assumptions in the Fuel Route/Dropped Load PSA were made in the development phase. They relate to each aspects of PSA, such as the identification of initiating events, the definition of success criteria, reliability data selection, containment performance, accident progression and so on. In addition, when the level of plant requirements specification limits the detail of the PSA, then appropriate assumptions were developed to allow completion of the PSA.

Starting from assumptions, sensitivity analyses have been performed. Among assumptions in the Fuel Route/Dropped Load PSA, key assumptions, which have comparatively large impact on the result, have been listed from the result of sensitivity analyse.

The key assumptions considered in the Fuel Route/Dropped Load PSA are listed below.

- It is assumed that AC power for the CCS is provided by the BBG. Operator action for the connection of the CCS to cask is credited as ___-HFE-CS-CA.
- Notes: The probability is conservatively assumed based on the similarity between this CCS connection to cask and FLSR connection to reactor building. The failure probability of AC power delivery from the BBG to CCS is assumed to be included in the HEP of ___-HFE-CS-CA because that failure probability smaller than the HEP.
- SGTS success criteria are based on the assumption of SGTS capacity. Sensitivity case considers when both two trains of SGTS are needed to filter the fission product.
- The number of cask movement, 25 /y is conservatively assumed.

25.13 PSA for Events Not Leading to a Core or Fuel Melt

25.13.1 Scope of Events Not Leading to a Core or Fuel Melt

This section provides the assessment of events that do not lead to a core or fuel melt, i.e. non-core damage (in this context core damage and fuel damage sequences lead to fuel melt) sequence groups and non-fuel damage sequence groups from the PSAs and those Non-Reactor faults not considered in the PSA models.

Core damage sequences from reactor faults, for both at Power and shutdown conditions, and fuel damage sequences from fuel route faults are treated in their respective PSA. In this assessment, non-core damage sequences and non-fuel damage sequences have been evaluated. These sequences, shown as the success sequences in the Event Trees in the PSA models, were categorised into a number of sequence groups, which were then taken forward to the Level 3 PSA. For non-reactor faults, the faults were related to the turbine systems and radwaste systems, which were not included in the PSA models. All systems were reviewed to identify the initiating events for the non-reactor faults.

When assigning representative fission product releases to each sequence group, the following assumptions have been made.

Reactor at Power PSA

Success sequences in the at power Level 1 PSA model have intact fuel or perforated fuel. In the case of intact fuel, the fission product release is assumed to be a fission product spike (Iodine spike) from the core to the reactor coolant. For perforated fuel cases, the source term is assessed after assessment of a perforated fuel release to the reactor coolant. Fission products subsequently leak through the containment boundary (for isolation events) or are assumed to be released from the non-isolated line (for non-isolation events).

Shutdown PSA

Success sequences in the Shutdown Level 1 PSA model have an intact core (with no fuel uncovery) with or without boiling of the coolant. In these cases, the fission product release is assumed to be from the water surrounding the fuel. The containment boundary is treated as always open for the purpose of maintenance even when the RPV and containment heads are on. The SGTS is conservatively not credited.

Fuel Route PSA (including SFP PSA)

Success sequences in the PSA model have intact fuel (no fuel uncovery in SFP PSA) or physically damaged (in this context physically damaged fuel does not melt) fuel in the dropped load case (Fuel Route PSA). In these cases, the fission product release is assumed to be from the fuel and/or the non-fuel material, such as the water surrounding the fuel.

The fission products are released to the Reactor Building. The fission products in the Reactor Building are expected to be released through the Main Stack. When the release to the environment through the Main Stack is considered, the availability of the SGTS is considered for the Fuel Route PSA. The SGTS is conservatively not credited for the SFP PSA success sequences.

Other faults including Turbine system and Radwaste system

Other faults not directly related to the Initiating Events described above, and not considered in the PSA models, were identified through the FMEA exercise for all systems in the UK ABWR [Ref-25.123]. Furthermore, the consequence analyses were performed for the bounding cases for these events. According to the results [Ref-25.23], the effective dose is significantly lower than 0.1 mSv (lower than the lowest dose band of 0.1 to 1 mSv in Target 8). In this case, a detailed frequency estimate is not necessary for the comparison to Target 8.

25.13.2 Interface with Level 3 PSA

25.13.2.1 Reactor and SFP PSAs

(1) IEAP PSA

Success sequences in the IEAP Level 1 PSA (Section 25.4) that do not lead to fuel melt, are categorised into 5 groups in the IEAP event sequence analysis. Each group is divided into: (i) no fuel failure sequences and (ii) fuel perforation sequences. Events with excessive clad oxidation and gap release, but not leading to fuel melt, are included in the fuel perforation sequence groups (see Section 25.4.3.2). In the fuel perforation sequences, pellet relocation is not expected. Sequences with pellet relocation are included in the release category frequencies, i.e. with the core melt sequences.

In total, the following ten sequence groups are identified for IEAP for the Level 3 PSA:

- **R : Success sequence by RHR operation (containment intact)**

  This group includes both LOCA and non LOCA sequences with successful RHR operation, including the failure to scram event (ATWS) and High-Energy Line Breaks (HELBs). These sequences are treated in the IEAP Level 1 PSA model, which includes Large LOCA with a pellet to clad gap release and successful ECCS.

  Approaches for other sequences in this group are described below:

  - Core power/flow oscillations have been analysed as ATWSI. The results show that there is no fuel damage in these events. The effect of an iodine spike release can be bounded by other DBA cases (e.g. LOCA). Considering the above characteristics, these events are included in this group because this event has no fuel perforation in the deterministic analysis [Chapter 24 of PCSR Rev. C] and radiological impact is similar to other RHR success sequences.
  - Control rod withdrawal accidents were analysed and no fuel failure was confirmed in the deterministic analysis [Chapter 24 of PCSR Rev. C]. In this accident, the control rod is recovered or the reactor is shutdown. The control rod withdrawal accident is included in this group because the radiological impact is expected to be similar to or smaller than this group, regardless of containment isolation.
  - HELB with successful isolation of containment and subsequent core water makeup and heat removal is also in this group, as it does not result in fuel perforation. The HELB dose evaluation is bounded by the Main Steam Line Break (MSLB), CUW Line Break and instrumentation line break evaluations.
  - Sequences with unsuccessful containment isolation in this group are assessed. When those sequences are assessed, large LOCA, failure to scram event and large core power/flow oscillations are considered.

- **R* : Success sequence by RHR operation (containment intact) with fuel perforation**

  This group includes both LOCA and non LOCA with successful RHR operation. Containment isolation is successful in this group. Fuel in this group is perforated in the deterministic analysis.

- **C : Success sequence by containment venting (containment intact)**

  In this group, the core is cooled by water makeup from an external water source and the containment is vented.

- **C* : Success sequence by containment venting (containment intact) with fuel perforation**

  In this group, the core is cooled by water makeup from an external water source and the containment is vented. Fuel in this group is perforated in the deterministic analysis.

- **CP : Success sequence with containment overpressure failure (with success of VSS)**

  In this group, the core is cooled by water makeup from an external water source. The containment venting is failed and containment is failed by overpressure. Water makeup by FLSS or Feed water continues, since the steam from containment does not affect equipment located outside the Reactor Building.

- **CP* : Success sequence with containment overpressure failure (with success of VSS) with fuel perforation**

  In this group, the core is cooled by water makeup from an external water source. The containment venting is failed and containment is failed by overpressure. Water makeup by FLSS or Feed water continues, since the steam from containment does not affect equipment located outside the Reactor Building. Fuel in this group is perforated in the deterministic analysis.

- **VP : Success sequence with containment failure (with VSS failure)**

  In this group, the core is cooled by water makeup from an external water source. The containment is failed by overpressure under VSS failure. Water makeup by FLSS or Feed water continues, since the steam from containment does not affect equipment located outside the Reactor Building.

- **VP* : Success sequence with containment failure (with VSS failure) with fuel perforation**

  In this group, the core is cooled by water makeup from an external water source. The containment is failed by overpressure under VSS failure. Water make up by FLSS or Feed water continues, since the steam from containment does not affect equipment located outside the Reactor Building. Fuel in this group is perforated in the deterministic analysis.

- **B : Success sequence with containment bypass**

  This group includes the following success sequences:

  - Interfacing System LOCA (ISLOCA)
  - Break Outside Containment (BOC) from IEAP PSA

  The frequency of this group was calculated.

- **B* : Success sequence with containment bypass with fuel perforation**

  CUW piping break with containment isolation unsuccessful is identified as the fuel perforation case with containment bypass in IEAP LOCA.

(2) Shutdown PSA

For the Shutdown PSA (Section 25.8), all success end states are divided into two sequence groups for Shutdown in the Level 3 PSA:

- **RS : Success sequence by RHR in shutdown mode**

  This group consists of success sequences by RHR in shutdown mode. In the Level 3 PSA, this represents both sequence group R and sequence group RS. The difference between group RS and group R is that the containment is not isolated and the reactor condition is 24 hours after shutdown for group RS.

  In the event of control rod withdrawal error, there is no fuel failure in the deterministic analysis [Ref-25.117]. Thus, control rod withdrawal error is in this group because the radiological impact is expected to be similar to or smaller than this group.

- **CS : Success sequence by water makeup in shutdown mode**

  This group has loss of decay heat from the reactor success sequences. In this group, the core is cooled by water makeup from an external water source.

(3) SFP PSA

Success end states which are not fuel damage sequences are categorised into two sequence groups:

- **PR : Success sequence by FPC or RHR in SFP**

  This group represents success sequences by FPC or RHR in SFP.

- **PM : Success sequence by water makeup in SFP**

  This group represents loss of decay heat from the SFP and the SFP is cooled by water makeup from an external water source.

25.13.2.2 Fuel Route PSA

All success sequences for the Fuel Route PSA (Section 25.12) are grouped into a fuel handling accident group and a heavy load (non fuel) drop group. For fuel handling accident groups (FS, FH), the fault which gives the largest impact in terms of offsite dose is the case involving irradiated fuel drop into the reactor core and it bounds other fuel handling accidents. For the heavy load drop group, three conditions have been assessed: cask drop onto fuel assemblies (HLPS, HLPF), cask drop onto cask pit or preparation pit (CDPS, CDPF), and cask drop in truck bay shaft (CDTS, CDTF). Each sub-group is subdivided into two sequence groups (i) a sequence group with SGTS unsuccessful and (ii) a sequence group with SGTS successful.

In total, the following eight sequence groups are identified for the Fuel Route for the Level 3 PSA:

- **FS : Fuel Handling Accident with SGTS successful**

  This group addresses Fuel Handling Accidents. Fuel assemblies physically damaged by the drop are evaluated. Fission products from the damaged fuel assemblies are released to the environment from the Reactor Building after filtration and hold-up through SGTS. The frequency of this group was calculated in the quantification notebook for Fuel Route PSA in Section 25.12. In the Fuel Route PSA, this group is identical to accident progression sequence/release category FD01.

- **FH : Fuel Handling Accident with SGTS unsuccessful**

  This group is Fuel Handling Accidents with SGTS unsuccessful. Fission products from the damaged fuel assemblies are released to environment from the Reactor Building. The frequency of this group was calculated in quantification for Fuel Route PSA [Ref-25.124]. In the Fuel Route PSA, this group consists of accident progression sequence/release category FD02 and FD03.

- **HLPS : Heavy Load (e.g. cask) drop into spent fuel pool with SGTS successful**

  This group is heavy load drop accidents into the spent fuel pool. The representative sequence is a cask drop onto the spent fuel storage rack. Fission products from the damaged fuel assemblies are released to environment from the Reactor Building after filtration and hold-up through SGTS. The frequency of this group was calculated in quantification for Fuel Route PSA [Ref-25.124]. In the Fuel Route PSA, this group is identical to accident progression sequence/release category CD-SFP01.

- **HLPF : Heavy Load (e.g. cask) drop into spent fuel pool with SGTS unsuccessful**

  This group is heavy load drop accident into spent fuel pool. The representative sequence is cask drop into spent fuel storage rack. Fission products from the damaged fuel assemblies are released to environment from the Reactor Building. The frequency of this group was calculated in quantification for Fuel Route PSA. In the Fuel Route PSA, this group consists of accident progression sequence/release category CD-SFP02 and CD-SFP03.

- **CDPS : Cask drop onto cask pit or preparation pit with SGTS successful**

  This group is cask drop accident onto cask pit or preparation pit. Fission products from the damaged fuel assemblies are released to environment from the Reactor Building after filtration and hold-up through SGTS. The frequency of this group was calculated in quantification for Fuel Route PSA. In the Fuel Route PSA [Ref-25.124], this group is identical to accident progression sequence/release category CD-CP01.

- **CDPF : Cask drop onto cask pit or preparation pit with SGTS unsuccessful**

  This group is cask drop accident onto cask pit or preparation pit. Fission products from the damaged fuel assemblies are released to environment from the Reactor Building. The frequency of this group was calculated in quantification for the Fuel Route PSA [Ref-25.124]. In the Fuel Route PSA, this group consists of accident progression sequence/release category CD-CP02 and CD-CP03.

- **CDTS : Cask drop onto truck bay shaft or LOOP with SGTS successful**

  This group is cask drop accidents onto the truck bay shaft and LOOP. Fission products from the damaged fuel assemblies are released to environment from the Reactor Building after filtration and hold-up through SGTS. The frequency of this group was calculated in the quantification for the Fuel Route PSA [Ref-25.124]. In the Fuel Route PSA, this group consists of accident progression sequence/release category CD02, CD05 and LOOP-DR03.

- **CDTF : Cask drop onto truck bay shaft or LOOP with SGTS unsuccessful**

  This group is cask drop accidents onto the truck bay shaft. Fission products from the damaged fuel assemblies are released to environment from the Reactor Building. The frequency of this group was calculated in the quantification for the Fuel Route PSA [Ref-25.124]. In addition, a LOOP during the cask handling operation scenario is added in this group. In the Fuel Route PSA, this group consists of accident progression sequence/release category CD03, CD04, CD06, CD07, LOOP-DR04 and LOOP-DR05.

25.13.2.3 Non-reactor faults

In the deterministic assessment of Non-reactor faults [Ref-25.23], all faults which are considered in the Level 3 PSA are grouped in the three fault groups listed below. The group is represented by the bounding fault in each group.

Other faults, which are not included in the listed groups, are screened out because of the small doses estimated for such events.

- **NT : Release/Leak from Turbine system**

  This group covers all leakage from the Turbine system faults. The bounding fault in this group is the Main Steam Line Break Accident [Ref-25.23].

- **NO : Off-gas system Rupture**

  This group is Off-Gas system (OG) system rupture faults. Fission products from the ruptured component are released to environment from the main stack through the building HVAC system.

- **NL : Leak from Radwaste system**

  This group covers all leakage from the Radwaste system faults. Fission products from the Radwaste system are released to environment from the building HVAC system.

25.13.3 Analysis for Level 3 PSA (Events not leading to fuel melt)

This section presents the Level 3 PSA analysis for the sequence/fault groups not leading to fuel melt. These consist of:

- PSA 'success' sequence groups, i.e. those sequence groups considered in the PSAs for the reactor at Power and Shutdown that do not lead to fuel melt,
- SFP and Fuel Route 'success' sequence groups, i.e. those sequence groups considered in the PSAs for the SFP and Fuel Route that do not lead to fuel melt, and
- Non-Reactor fault groups, i.e. those not considered in the PSAs that do not lead to fuel melt.

The PSA 'success' sequence groups and non-PSA fault group descriptions and their representative frequencies are given in Table 25.13.3-1, Table 25.13.3-2 and Table 25.13.3-3. PSA 'success' sequence groups, non-PSA fault groups and their summated frequencies were assessed.

Twelve sequence groups are brought forward from the IE at Power and IE at Shutdown Level 1 PSAs, and are summarised in Table 25.13.3-1. Three of these sequence groups are screened out on the basis of low consequences in sub-section 25.13.4.1 below. Therefore, their frequencies are not carried forward into the Level 3 PSA and are not given in Table 25.13.3-1. The summated frequencies of the PSA 'success' sequence groups taken forward to the Level 3 PSA is 2.22E-03 /y.

Ten additional sequence groups, brought forward from the SFP IE Level 1 PSA and Fuel Route PSA are considered together. The two sequence groups brought forward forms the SFP IE Level 1 PSA are included in Table 25.13.3-2. The summated frequencies of the SFP and Fuel Route PSA 'success' sequence groups taken forward to the Level 3 PSA is 1.32E-01 /y.

Four fault groups and their summated frequencies, brought forward from the DBA, are summarised in Table 25.13.3-3. Three of these fault groups are screened out on the basis of low consequences in subsection 25.13.4.1 below. Therefore, their summated frequency is not carried forward into the Level 3 PSA and is not given in Table 25.13.3-2. The summated frequency of the non-PSA fault groups taken forward to the Level 3 PSA is 1.0E-05 /y.

The representative doses for the PSA ‘success’ sequence groups and non-PSA fault groups, for assessment against the facility dose bands were assessed. These doses are calculated using the more conservative methodology of DBA. At this time, no allowance has been made for the more realistic source term, dispersion and dose uptake characteristics appropriate for assessment against Target 8. However, should this be identified as a significant conservatism in the PSA, this may be done at the site licensing stage.

The probabilistic consequences calculations have been performed using the same bounding radionuclide releases as the deterministic dose calculations but use the same methodology as the other probabilistic consequences calculations (see Section 25.6.4).

Also given in Table 25.13.3-1, Table 25.13.3-2 and Table 25.13.3-3 are the Level 3 PSA case identifiers used to represent the conditional consequences in the Level 3 PSA.

Table 25.13.3-1 PSA 'Success' Sequence Groups for the Reactor at Power and Shutdown, Summated Frequencies and Representative Level 3 PSA Cases

PSA 'success' sequence groups	Frequency taken forward (/y)	Comment / Level 3 PSA case
R : Success sequence by RHR operation (containment intact)	n/a	Screened out on low consequence
C : Success sequence by containment venting (containment intact)	6.63E-04	C
CP : Success sequence with containment overpressure failure (with success of VSS)	1.36E-03	CP
VP : Success sequence with containment failure (with VSS failure)	8.19E-07	VP
B : Success sequence with containment bypass	1.51E-05	B
R* : Success sequence by RHR operation (containment intact) with fuel perforation	1.78E-04	R*
C* : Success sequence by containment venting (containment intact) with fuel perforation	5.39E-06	C*
CP* : Success sequence with containment overpressure failure (with success of VSS) with fuel perforation	1.21E-07	CP*
VP* : Success sequence with containment failure (with VSS failure) with fuel perforation	3.66E-08	VP*
B* : Success sequence with containment bypass with fuel perforation	3.12E-09	B*
RS : Success sequence by RHR in shutdown mode	n/a	Screened out on low consequence
CS :Success sequence by water makeup in shutdown mode	n/a	Screened out on low consequence
Total frequency taken forward (/y):	**2.22E-03**	

Table 25.13.3-2 PSA 'Success' Sequence Groups for the SFP and Fuel Route, Summated Frequencies and Representative Level 3 PSA Cases

Non-PSA fault groups	Frequency (/y)	Comment / Level 3 PSA case
PM : Success sequence by water makeup in SFP	1.32E-01	PM
PR : Success sequence by FPC or RHR in SFP	n/a	Screened out on low consequence
FS : Fuel Handling Accident with SGTS successful	3.19E-04	FS
FH : Fuel Handling Accident with SGTS unsuccessful	6.51E-05	FH
HLPS : Heavy Load (e.g. cask) drop into spent fuel pool with SGTS successful	1.43E-08	HLPS
HLPF : Heavy Load (e.g. cask) drop into spent fuel pool with SGTS unsuccessful	3.93E-10	HLPF
CDPS : Cask drop onto cask pit or preparation pit with SGTS successful	2.07E-05	CDPS
CDPF : Cask drop onto cask pit or preparation pit with SGTS unsuccessful	7.60E-07	CDPF
CDTS : Cask drop onto truck bay shaft or LOOP with SGTS successful	2.67E-08	CDTS
CDTF : Cask drop onto truck bay shaft or LOOP with SGTS unsuccessful	4.85E-09	CDTF
Total frequency taken forward (/y):	**1.32E-01**	

Table 25.13.3-3 Other (non-PSA) Fault Groups, Summated Frequencies and Representative Level 3 PSA Cases

Non-PSA fault groups	Frequency (/y)	Comment / Level 3 PSA case
NO1 : Off-gas System Failure unmitigated case	1.00E-05	NO: Case 1 + NO: H/U Rupture
NO2 : Off-gas System Failure mitigated case	n/a	Screened out on low consequence
NL : Leak from Radwaste system	n/a	Screened out on low consequence
NT : Release/Leak from Turbine system	n/a	Screened out on low consequence
Total frequency taken forward (/y):	**1.00E-05**	

25.13.4 Results of Level 3 PSA (Events not leading to a core or fuel melt)

25.13.4.1 Facility Dose Bands

Table 25.13.4-1 and Table 25.13.4-2 show the facility dose band allocations, which are determined from the maximum long term dose (cloud gamma, inhalation, ingestion and long term ground gamma) for each case. Where the sum of the cloud gamma, inhalation and long term ground gamma dose is > 1 mSv, the ingestion dose is not included and the frequency is considered to contribute to the frequency of food bans.

Where the dose is <0.01 mSv, the sequence/fault group is screened out from further assessments against Target 7. This is the level of dose usually considered as trivial for fault assessment.

It can be seen that:

- The following sequence/fault groups fall well below the threshold of 0.01 mSv and are not taken forward into the assessment against Target 7:
  - R : Success sequence by RHR operation (containment intact),
  - RS : Success sequence by RHR in shutdown mode,
  - CS : Success sequence by water makeup in shutdown mode,
  - PR : Success sequence by FPC or RHR in SFP,
  - NO2 : Off-gas System Failure mitigated case,
  - NL : Leak from Radwaste system, and
  - NT : Release/Leak from Turbine system,
  - FS : Fuel Handling Accident with SGTS successful.
- The following sequence/fault groups fall below the Target 8 lower threshold of 0.1 mSv but are taken forward into the assessment against Target 7:
  - C : Success sequence by containment venting (containment intact),
  - VP : Success sequence with containment failure (with VSS failure),
  - B : Success sequence with containment bypass, and
  - PM : Success sequence by water makeup in SFP.
- The following sequence/fault groups fall above the Target 8 lower threshold of 0.1 mSv and are in the facility dose band 0.0001 to 0.001 Sv:
  - CP : Success sequence with containment overpressure failure (with success of VSS),
  - NO1 : Off-gas System Failure unmitigated case,
  - CDPS : Cask drop onto cask pit or preparation pit with SGTS successful,
  - CDTS : Cask drop onto truck bay shaft or LOOP with SGTS successful, and
  - HLPS : Heavy Load (e.g. cask) drop into spent fuel pool with SGTS successful.
- The following sequence/fault groups fall in the facility dose band 0.001 to 0.01 Sv:
  - FH : Fuel Handling Accident with SGTS unsuccessful, and
  - R* : Success sequence by RHR operation (containment intact) with fuel perforation.
- The following sequence/fault groups fall in the facility dose band 0.01 to 0.1 Sv:

    - C* : Success sequence by containment venting (containment intact) with fuel perforation,
    - HLPF : Heavy Load (e.g. cask) drop into spent fuel pool with SGTS unsuccessful, and
    - CDPF : Cask drop onto cask pit or preparation pit with SGTS unsuccessful.
- The remaining sequence/fault groups fall in the >1 Sv facility dose band:
    - B* : Success sequence with containment bypass with fuel perforation,
    - CP* : Success sequence with containment overpressure failure (with success of VSS) with fuel perforation,
    - VP* : Success sequence with containment failure (with VSS failure) with fuel perforation, and
    - CDTF : Cask drop onto truck bay shaft or LOOP with SGTS unsuccessful.

Table 25.13.4-1 Allocation to Facility Dose Bands for PSA 'Success' Sequence Groups

PSA 'success' sequence groups	Facility Dose Band
R : Success sequence by RHR operation (containment intact) with fuel perforation	Screened out on low consequence
C : Success sequence by containment venting (containment intact)	<0.0001 Sv
CP : Success sequence with containment overpressure failure (with success of VSS)	0.0001 to 0.001 Sv
VP : Success sequence with containment failure (with VSS failure)	<0.0001 Sv
B : Success sequence with containment bypass	<0.0001 Sv
R* : Success sequence by RHR operation (containment intact) with fuel perforation	0.001 to 0.01 Sv
C* : Success sequence by containment venting (containment intact) with fuel perforation	0.01 to 0.1 Sv
CP* : Success sequence with containment overpressure failure (with success of VSS) with fuel perforation	>1 Sv
VP* : Success sequence with containment failure (with VSS failure) with fuel perforation	>1 Sv
B* : Success sequence with containment bypass with fuel perforation	>1 Sv

Note (1) – doses assume unrestricted ingestion dose calculated using 'Top Two' methodology.

Note (2) – ingestion doses are limited by food bans.

Table 25.13.4-2 Allocation to Facility Dose Bands for Other (non-PSA) Fault Groups Not Leading to Fuel Melt

Non-PSA fault groups	Facility Dose Band
RS : Success sequence by RHR in shutdown mode	Screened out on low consequence
CS : Success sequence by water makeup in shutdown mode	Screened out on low consequence
PM : Success sequence by water makeup in SFP	<0.0001 Sv
PR : Success sequence by FPC or RHR in SFP	Screened out on low consequence
FS : Fuel Handling Accident with SGTS successful	Screened out on low consequence
FH : Fuel Handling Accident with SGTS unsuccessful	0.001 to 0.01 Sv
HLPS : Heavy Load (e.g. cask) drop into spent fuel pool with SGTS successful	0.0001 to 0.001 Sv
HLPF : Heavy Load (e.g. cask) drop into spent fuel pool with SGTS unsuccessful	0.01 to 0.1 Sv
CDPS : Cask drop onto cask pit or preparation pit with SGTS successful	0.0001 to 0.001 Sv
CDPF : Cask drop onto cask pit or preparation pit with SGTS unsuccessful	0.01 to 0.1 Sv
CDTS : Cask drop onto truck bay shaft or LOOP with SGTS successful	0.0001 to 0.001 Sv
CDTF : Cask drop onto truck bay shaft or LOOP with SGTS unsuccessful	>1 Sv
NO1 : Off-gas System Failure unmitigated case	0.0001 to 0.001 Sv
NO2 : Off-gas System Failure mitigated case	Screened out on low consequence
NL : Leak from Radwaste system	Screened out on low consequence
NT : Release/Leak from Turbine system	Screened out on low consequence

Note (1) – doses assume unrestricted ingestion dose calculated using 'Top Two' methodology.

Note (2) – ingestion doses are limited by food bans.

Table 25.13.4-3 presents the assessment against facility dose bands for the events not leading to fuel melt. It can be seen that there is a significant margin to the BSO for all dose bands:

- The total contribution to the 0.0001 to 0.001 Sv (0.1 to 1 mSv) facility dose band is 1.39E-03 /y. This is equivalent to 13.9 percent of the BSO:

- CP : Success sequence with containment overpressure failure (with success of VSS), contributes 1.36E-03 /y or 97.8 percent of the total band frequency.
- CDPS : Cask drop onto cask pit or preparation pit with SGTS successful contributes 2.07E-05 /y or 1.5 percent of the total band frequency.

- The total contribution to the 0.001 to 0.01 Sv (1 to 10 mSv) facility dose band is 2.43E-04 /y. This is equivalent to 24.3 percent of the BSO:
  - R* : Success sequence by RHR operation (containment intact) with fuel perforation, contributes 1.78E-04 /y or 73.2 percent of the total band frequency.
  - FH : Success sequence by RHR operation (containment intact) with fuel perforation, contributes 6.51E-05 /y or 26.8 percent of the total band frequency.
- The total contribution to the 0.01 to 0.1 Sv (10 to 100 mSv) facility dose band is 6.15E-06 /y. This is equivalent to 6.2 percent of the BSO:
  - C* : Success sequence by containment venting (containment intact) with fuel perforation, contributes 5.39E-06 /y or 87.6 percent of the total band frequency.
  - CDPF : Cask drop onto cask pit or preparation pit with SGTS unsuccessful, contributes 7.60E-07 /y or 12.4 percent of the total band frequency.
- The total contribution to the >1 Sv (>1,000 mSv) facility dose band is 1.66E-07 /y. This is equivalent to 16.6 percent of the BSO. Four sequence/fault groups contribute to this frequency:
  - CP* : Success sequence with containment overpressure failure (with success of VSS) with fuel perforation, contributes 1.21E-07 /y or 72.9 percent of the total band frequency.
  - VP* : Success sequence with containment failure (with VSS failure) with fuel perforation, contributes 3.66E-08 /y or 22.1 percent of the total band frequency.
  - CDTF : Cask drop onto truck bay shaft or LOOP with SGTS unsuccessful, contributes 4.85E-09 /y or 2.9 percent of the total band frequency.
  - B* : Success sequence with containment bypass with fuel perforation, contributes 3.12E-09 /y or 1.9 percent of the total band frequency.

Table 25.13.4-3 Assessment against Facility Dose Bands (Target 8) for Sequence / Fault Groups Not Leading to Fuel Melt

Effective dose (Sv)	Bounding Sequence / Bounding Fault	Frequency (/y)	Percentage of BSO	BSO	BSL
0.0001 to 0.001	CP, HLPS, CDPS, CDTS, NO1 + H/U Rupture	1.39E-03	13.9 %	1.0E-2	1.0E+0
0.001 to 0.01	R*, FH	2.43E-04	24.3 %	1.0E-3	1.0E-1
0.01 to 0.1	C*,HLPF, CDPF	6.15E-06	6.2 %	1.0E-4	1.0E-2
0.1 to 1		0.00E+00	0.0 %	1.0E-5	1.0E-3
> 1	CP*, VP*, B*, CDTF	1.66E-07	16.6 %	1.0E-6	1.0E-4
Summated frequency of dose bans /y		1.64E-03			

25.13.4.2 Assessment against Target 7

A smaller number of Level 3 PSA cases are analysed using probabilistic methods, as a result of subsuming several fault groups into a single bounding Level 3 PSA case. Conditional individual risks for the Level 3 PSA cases are given in [Ref-25.1] for each of the sequence/fault groups, at distances of 400 m, 1,000 m and 1,500 m from the site to illustrate how the conditional individual risk varies with distance over the range of likely offsite locations with continuous occupancy. The conditional individual risk results in this table are calculated using a single release phase and minimal protective actions.

The conditional individual risks given in [Ref-25.1] are due to late (stochastic) health effects leading to a fatality. The conditional risk of early (deterministic) health effects leading to a fatality is zero in all cases.

The individual risk for each sequence/fault group is calculated as the product of the conditional individual risk at the three distances given in [Ref-25.1] and the representative frequency given in Table 25.13.3-1, Table 25.13.3-2 and Table 25.13.3-3. The contribution of each of the sequence/fault groups to the overall risk from events not leading to fuel melt is given in Table 25.13.4-4 and Figure 25.13.4-1.

The summated individual risk at 1 km from events not leading to fuel melt is 2.05E-09 /y or 0.21 percent of the BSO. Two sequence/ groups contribute almost all of the individual risk:

- PM : Success sequence by water makeup in SFP, contributes 1.74E-09 /y or 84.89 percent of the total
- R* : Success sequence by RHR operation (containment intact) with fuel perforation, contributes 1.69E-10 /y or 8.22 percent of the total

Table 25.13.4-4 Individual Risk of Fatality Close to the Site (Target 7) for Sequence / Fault Groups Not Leading to Fuel Melt

Release Category	Release Category Frequency (/y)	Individual risk of fatal health effects (/y)			Contribution to Total Risk at 1 km
		400 m	1,000 m	1,500 m	
C	6.63E-04	2.83E-11	8.40E-12	5.01E-12	0.41 %
CP	1.36E-03	5.80E-11	1.72E-11	1.03E-11	0.84 %
VP	8.19E-07	3.49E-14	1.04E-14	6.19E-15	<0.01 %
B	1.51E-05	9.01E-13	2.68E-13	1.60E-13	0.01 %
R*	1.78E-04	4.93E-10	1.69E-10	1.05E-10	8.22 %
C*	5.39E-06	1.16E-10	4.53E-11	3.10E-11	2.21 %
CP*	1.21E-07	7.69E-11	2.34E-11	1.42E-11	1.14 %
VP*	3.66E-08	1.39E-11	4.29E-12	2.59E-12	0.21 %
B*	3.12E-09	5.66E-13	1.74E-13	1.07E-13	0.01 %
PM	1.32E-01	5.85E-09	1.74E-09	1.03E-09	84.89 %
FS	3.19E-04	1.54E-11	4.75E-12	2.86E-12	0.23 %
FH	6.51E-05	3.14E-12	9.70E-13	5.83E-13	0.05 %
HLPS	1.43E-08	6.47E-14	2.21E-14	1.38E-14	<0.01 %
HLPF	3.93E-10	1.78E-15	6.07E-16	3.79E-16	<0.01 %
CDPS	2.07E-05	9.36E-11	3.20E-11	2.00E-11	1.56 %
CDPF	7.60E-07	3.44E-12	1.17E-12	7.34E-13	0.06 %
CDTS	2.67E-08	1.12E-11	2.90E-12	1.62E-12	0.14 %
CDTF	4.85E-09	2.03E-12	5.27E-13	2.94E-13	0.03 %
NO1 (unmitigated) + H/U Rupture	1.00E-05	1.29E-14	3.90E-15	2.30E-15	<0.01 %
Total individual risk (/y):		**6.76E-09**	**2.05E-09**	**1.23E-09**	
Total as % of BSO		**0.68 %**	**0.21 %**	**0.12 %**	
BSO		1.00E-06			
BSL		1.00E-04			

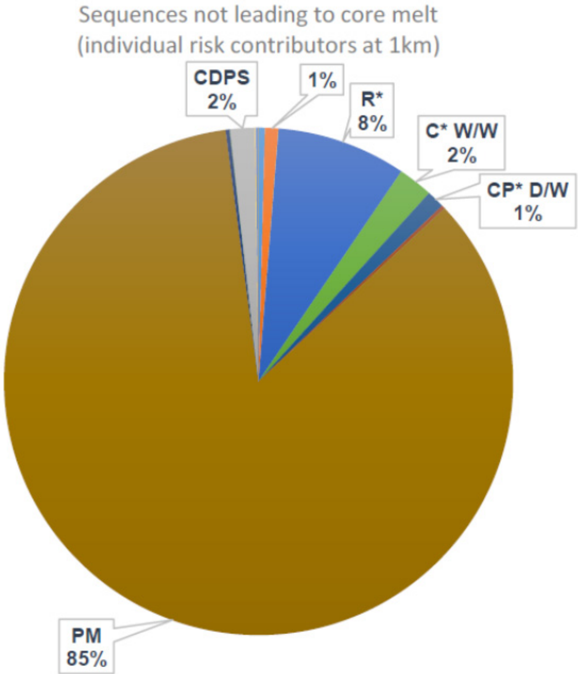


Figure 25.13.4-1 Contribution to the Individual Risk at 1 km for Sequence / Fault Groups Not Leading to Fuel Melt

25.13.4.3 Assessment against Target 9

The mean numbers of notional late fatalities for each of the Level 3 PSA cases are given in [Ref-25.1], assuming minimal offsite protective actions. These are broken down into the mean number of early (deterministic) fatal health effects, the mean number of notional late (stochastic) fatal health effects in the population, and the conditional probability of exceeding 100 notional late fatalities in the population.

The conditional risk of early (deterministic) health effects leading to a fatality is zero in all cases. The mean number of notional late (stochastic) health effects leading to a fatality is <100 in all cases.

For consistency with the methodology developed for events leading to core melt, the mean values of fatalities (sum of mean number of early fatal health effects and number of notional late health effects in the population) are used for comparison against Target 9 within GDA. If the mean value is >100 then all the frequency is allocated as above the threshold and, similarly, if the mean value is <100 then none of the frequency is allocated above the threshold.

The conditional probability of exceeding 100 notional late fatalities is non-zero for three Level 3 PSA cases: CP*, VP* and B*. It is considered that the conditional probability of exceeding the Target 9 threshold may be sensitive to the minimal protective action assumption, as a significant contribution comes from the collective ingestion dose from eating food at levels below the CFILs (which could be due to large numbers of trivial individual doses).

The current direct comparison approach is considered more robust as only a small number of Level 3 PSA cases are considered to be 'borderline' (P2, CP*, VP* and B*) and the direct comparison approach leads to slightly more conservative results in the overall summation for Target 9 (see Section 25.15.4).

Therefore, all events not leading to fuel melt are considered to be below the Target 9 threshold and this initiating event group does not contribute to the frequency of exceeding the Target 9 threshold.

As mentioned above, all cases are currently considered to be below the Target 9 threshold. However, based on the current conditional probabilities calculated with minimal protective actions, the maximum frequency that could be above the Target 9 threshold for this group is 2.84E-08 /y (see Section 25.15.4).

25.13.5 Insights from Assessment

The summated individual risk at 1 km from events not leading to fuel melt is 2.05E-09 /y or 0.21 percent of the BSO. Two sequence/ groups contribute almost all of the individual risk:

- PM : Success sequence by water makeup in SFP, contributes 1.74E-09 /y or 84.89 percent of the total
- R* : Success sequence by RHR operation (containment intact) with fuel perforation, contributes 1.69E-10 /y or 8.22 percent of the total

All cases are currently considered to be below the Target 9 threshold. However, based on the current conditional probabilities calculated with minimal protective actions, the maximum frequency that could be assigned as above the Target 9 threshold for this group is 2.84E-08 /y.

The frequencies used for the assessments are based on the internal events PSAs. It was concluded that the BSO of Target 7 and BSOs of Target 8 (except for > 1,000 mSv dose band) would not be challenged (i.e., sufficient margins to BSOs retained) even if the low dose sequences caused by internal hazards and external hazards additionally contribute to the frequencies of each release group [Ref-25.122]. It was also concluded that the dose band frequency above 1,000 mSv (including Level 2 PSA end states) would be only marginally increased by additionally considering the CDTF, CP*, VP* and B* group frequencies from internal hazards and external hazards [Ref-25.122].

25.13.6 Key Assumptions and Study Limitations

Key assumptions used in the sequence frequency evaluations are listed below.

- Accident mitigation functions that have been credited in IEAP PSA, shutdown PSA and SFP PSA, such as FLSS, containment venting, etc., are also credited here.
- For a simplified assessment, the frequency of success sequences can be assessed to be the same as the IE frequency, because CDF and FDF are sufficiently small in UK ABWR and conditional success probability is approximately 1.
- The frequencies used for the assessments are based on the internal events PSAs. Additional risk from internal hazards and external hazards were semi-quantitatively addressed.

The exposure pathways, used as the basis of the conditional risk calculations are:

- The external dose received during the plume passage (cloud gamma),
- The committed dose due to inhalation during the plume passage,
- The long term external dose from radionuclides deposited on the ground (groundshine),
- The dose due to activity deposited on skin,
- The dose due to ingestion of contaminated foodstuffs (with food bans implemented should the EU CFILs be exceeded), and
- The dose due to long term resuspension of activity deposited on the ground.

Where the dose is <0.01 mSv, the sequence/fault group is screened out from further assessments against Target 7. This is the level of dose usually considered as trivial for fault assessments.

25.14 Worker Risk Assessment

25.14.1 Scope of Worker Risk Assessment

This section presents the approach and results of a preliminary assessment of potential radiological risk to a person on site against the NSEDP Numerical Targets [Ref-25.58] (equivalent to the Numerical Targets 5 and 6 set out in ONR SAPs), performed to demonstrate that the UK ABWR design can meet these targets.

The numerical risk targets in the ONR SAPs require:

- assessment of the summated individual risk of death from faults/accidents for any person on the site (Target 5), and
- assessment for each single fault against frequency/worker dose targets (Target 6).

The assessment has been based on preliminary assumptions. The worker dose estimates used in the risk calculations have not been explicitly calculated, but have largely been taken from existing design basis analysis (PCSR Chapter 24).

The assessment is limited to potential radiological risk to workers on site due to fault/accident conditions. Potential risks due to conventional hazards are not included. Potential risks associated with recovery and clean-up efforts are also not covered.

The detailed information is described in [Ref-25.131].

25.14.2 Methodology for the UK ABWR Worker Risk Assessment

The general principles that apply to the assessments for both Target 5 and Target 6 are listed below.

Fault identification

The PSA for the UK ABWR brings together a very large number of potential fault sequences considered to affect the plant. It is necessary to identify the faults to be considered in the worker risk assessment. Thus, there is a requirement to systematically identify fault/accident scenarios which could result in a significant effective dose to workers on the site (due to direct radiation and/or dose due to a release of radioactivity) and assess their consequences. The following sources need to be considered when deriving the list of representative faults for the worker risk assessment:

(1) The fault schedule, which is the definitive listing of all initiating events considered in the fault analysis,

(2) The characteristic faults considered in the DBA (and BDBA) for assessment against Target 4 and Target 8,

(3) Representative fault groups generated from the success sequences in the Level 1 PSA,

(4) The Release Categories generated from the Level 2 PSA,

(5) Faults identified from the Level 1 PSA (3) and Level 2 PSA (4) where an operator action is claimed to terminate the fault. In this case the dose due to performing the required action will be calculated in addition to the dose to a worker local to the fault, and

(6) Any other fault conditions identified as leading to unanticipated worker dose that are not considered in (2) or (3) above.

Frequency of Occurrence

Faults considered in the assessment against the probabilistic criteria of Target 6 and Target 5 is allocated a frequency of occurrence. As most of the representative faults are considered in some way in the Level 1 or Level 2 PSA, a corresponding frequency of occurrence can be derived. There may be some faults where there is not a straightforward way to derive a frequency so, in these cases, an estimate of the frequency of occurrence is made using engineering judgment.
Faults screened out of the assessment on the basis of low consequence do not need to be allocated a frequency of occurrence.

Dose Assessment Principles

The general principles applied to the dose assessments are:

(1) Only situations where there is a credible exposure path for workers are considered.

(2) Faults considered to result in worker doses < 0.1 mSv, based on simple conservative criteria, may be screened out of the assessment.

(3) The dose to a worker at the location where the dose is likely to be greatest will be calculated for all faults.

(4) The dose to workers in other locations with high occupancy, e.g. workers in the Main Control Room (MCR), may also be considered for some faults to support the assessment against Target 5 for important 'occupational area' groups.

(5) Where possible, the dose will be calculated using realistic values for source terms and other parameters which may influence the calculated dose. Where significant uncertainties exist, a more bounding assessment may be performed.

(6) Doses to workers from actions claimed (as required to terminate the fault) will be considered but worker doses from recovery and clean-up actions will not be considered.

Calculation of worker dose

The calculation of the radiological consequence at the exposed worker location from any fault considers the following:

(1) The location of the worker relative to any contained source and/or any activity release,

(2) The duration of the exposure, this should be a reasonably bounding estimate,

(3) The source term, which consist of either (or both) of two types, namely

- a contained source(s) at a location important to the worker dose assessment, and
- an uncontained source(s) of activity released from the plant/system.

Approach for assessment against Target 6

Following the general principles described above, the fault is allocated to the worker dose band that is representative of the most exposed worker, i.e. for the dose local to the faults or for the highest dose where alternate locations are considered.

Faults resulting in doses of < 2 mSv to the most exposed worker are screened from the Target 6 assessment as below the lowest dose assessment level.

For each fault considered to result in a worker dose > 2 mSv:

- The frequency of occurrence of the fault is compared against the BSO and BSL for the relevant worker dose band.
- When all faults have been individually assessed, the faults representing the greatest contributions to each worker dose band are identified. Any individual faults having a frequency above or close to the BSL will be identified for further consideration.
- For the faults identified in the previous step, a review is performed to identify any significant conservatism in the worker dose assessment, which may be removed and a lower dose calculated. In this case, the impact on the assessment against Target 6 is noted.

Approach for assessment against Target 5

The assessment against Target 5 use the same worker dose calculations as the comparison against Target 6. Using this data, the summated frequencies of faults allocated to each worker dose band are calculated for each of the main operational areas on the site, i.e. areas for which Occupancy Factors are defined.

A conditional worker risk is calculated for the representative faults in the worker risk assessment, using an appropriate dose/risk conversion factor. Alternatively, a single conditional risk value may be used to represent a range of faults, for example each dose band considered in Target 6.

The frequencies used in assessments against Target 5 are predicted frequencies of exposure. These differ from the fault frequencies used in the Target 6 assessment, which are frequencies of occurrence of the faults themselves. Due to normal working patterns, it would be too conservative to assume any individual can be at risk from all potential accidents for all of the time.

A ‘generic worker’ is considered as the main worker group for the assessment against Target 5. This worker group could potentially be exposed to the consequences of any accident within any building on site. However, Occupancy Factors are used to approximately express:

- the proportion of time the generic worker group will spend on-site per year, and
- when on-site, the average time spent in different buildings where faults can occur, as well as a proportion of time in non-specified ‘on-site’ areas external to the main buildings.

The Occupancy Factors, needed to perform the summated worker risk calculation for GDA are based on preliminary assumptions.

The summated risk to members of the ‘generic worker’ from faults is calculated using the components described above:

- frequency of occurrence of faults (and the operational area where they occur),
- the dose band which represents the most exposed worker,
- the conditional risk for each fault group or dose band, and
- the ‘generic worker’ Occupancy Factors for the operational areas.

Specifying additional worker groups may be useful in identifying whether the risk to members of a smaller group of workers could be greater than for members of the ‘generic worker’ group. Typically such groups may be dedicated to the Main Control Room, fuel handling operations or waste processing operations. In this case the worker risk calculation:

- only considers the faults directly affecting members of each smaller ‘operational area’ group,
- assumes that all the time the worker spends on-site is within the specified ‘operational area’.

25.14.3 Illustrative Results

Potential worker exposure situations have been identified in various deterministic assessments for a wide range of initiating events [Ref-25.23] [Ref-25.117] [Ref-25.128]. The exposure situations associated with the following fault types were considered for the purpose of this assessment:

- Station Blackout (SBO),
- Internal Events at Power,
- Internal Events at Shutdown,
- Non-Reactor events,
- SFP and Fuel Route events,
- Beyond Design Basis events,
- Preliminary assessment of dropped Dry Fuel Store canister,
- Preliminary assessment of Fuel Route and Dropped Load PSA,
- Preliminary assessment of exposure scenarios for Severe Accidents, and
- Worker Dose Evaluation Modelling.

Based on the general principles and approach summarised above, the results of the preliminary worker risk assessment are summarised as follows.

Assessment against the risk targets of NT.1 Target 5

The highest risk to a worker was calculated as lower than the BSO for Target 5 using the representative frequencies as shown in Table 25.14.3-1. The highest risk worker group in this preliminary assessment is the ‘involved’ worker, with 4.33E-07 /y, i.e. about 43 percent of the BSO for Target 5. This is largely attributable to the high consequence dropped load events assigned to the highest Target 6 dose band for ‘involved’ workers:

Assessment against the dose bands of NT.1 Target 6

Significant exposure situations (shown in Table 25.14.3-2) have been identified across the four Target 6 dose bands. There is only one fault (cask drop fault, CD-CP01) that could lead to a significant exposure situation that has a frequency higher than 1 percent of the BSL (i.e. higher than the BSO frequency). All the other faults in the top two dose bands are assigned frequencies less than the BSO. Only two fuel drop events allocated a frequency more than 0.1 percent of BSL (> 10 percent of the BSO) It is noted that none of the fault frequencies were modified to account for the reliability of backup systems which would prevent or mitigate the worker exposure situations following the faults, as described in the deterministic analysis.

Table 25.14.3-1 Summary of the Assessment against Target 5 Using Representative Frequencies

Contributors	Involved Worker	% of Total	MCR Worker	% of Total	Outside Worker	% of Total
Cask drop situations	1.46E-07	33.64 %	1.87E-10	38.51 %	9.35E-09	4.54 %
Severe Accidents	0	0	2.32E-10	47.89 %	3.52E-08	17.10 %
Others	2.88E-07	66.36 %	6.60E-11	13.60 %	1.61E-07	78.36 %
Total (all contributors) **% of BSO**	**4.33E-07** **43 %**		**4.85E-10** **0 %**		**2.06E-07** **21 %**	

Table 25.14.3-2 Identified Exposure Situations for the Comparison against Target 6

Target 6 dose band	BSL	Event Type	Fault Description	Allocated Frequency (/y)
>2,000	1 E-04	Cask Drop onto Cask Pit or Preparation Out	CD-CP02	5.55E-07
			CD-CP03	2.05E-07
		SA - Internal events	Sum of all release categories from internal events PSAs (excluding internal and external hazards initiators) and SFP PSAs	3.52E-07
200 to 2,000	1 E-03	Cask Drop onto Cask Pit or Preparation Out	CD-CP01	2.07E-05
20 to 200	1 E-02	Non-Reactor Faults	Fuel assembly failure due to dropped load	1 E-04
		Fuel drop and collision	Fuel drop into core during irradiated fuel handling between SFP rack and the core	1 E-04
		Drop of Heavy Equipment	RIP impeller shaft into the core	1 E-04
		Fuel Drop	FD02	3.93E-05
		Fuel Drop	FD03	2.58E-05
2 to 20	1 E-01	Internal Events during Shutdown	Loss of operating RHR with loss of same division of ECCS	1 E-03
		Internal Events during Shutdown	Drain down due to valve failure within operating RHR	1 E-03
		CCF of C&I systems	Inadvertent start-up A2 (FLSS) injection system	1E-03
		Radwaste system leak or failure	Off-Gas Treatment system failure (OG system rupture)	1E-03
			Spread of containment due to maintenance failure	1E-03
		Fuel Drop	FD01	3.19E-04

25.15 Summary of PSA Results and Key Insights

25.15.1 Introduction

Detailed PSA evaluations have been undertaken for all Plant Operation States and Initiating Events. PSA quantification results were obtained from the following UK ABWR PSAs:

- Internal Initiating Events at Power,
- Internal Initiating Events during Shutdown,
- Internal Initiating Events for the Spent Fuel Pool,
- Internal Fire and Flooding for at Power, Shutdown and the Spent Fuel Pool, and other Internal Hazard (Turbine Missile),
- Seismic Hazard for at Power, Shutdown and the Spent Fuel Pool, and other External Hazards (Tornado Missile, Accidental Aircraft Impact)
- Fuel Route, and
- Non Reactor Faults.

The individual results and insights from these PSAs are presented above in Sections 25.7, 8, 9, 10, 11, and 12 respectively.

As noted in the above sections, the GDA UK ABWR PSAs were performed using the design information that was available at the time analysis was performed. This by necessity has resulted in conservatism being made in the assessments. This is particularly true in the case of the hazards assessments where some aspects of the detailed design, such as cable layouts, are not finalised. Ideally, integration of the various assessments should be undertaken with an equal degree of conservatism being shared by all the assessments. However, this is not the case. The integration of the Hazards PSA results with those of the Internal Events PSA at this stage must therefore be considered is of limited value. Nonetheless an initial model integration has been performed to derive the overall risk of core/fuel damage and large release. In addition, system and component importance measures have been generated to provide additional insights into the robustness and adequacy of the UK ABWR design. The finding of these results and the insights gleaned were used to guide the ALARP assessment.

25.15.2 Summary of Level 1PSA Results

The UK ABWR PSA utilises Core Damage Frequency (CDF) and Fuel Damage Frequency (FDF) as interim measures of safety in the development of the UK ABWR design. As both provide a measure of potential damage done to fuel rods; CDF is used to measure damage in the core, while FDF is used to measure damage in the spent fuel pool. FDF and CDF provide insights into the success of the plant design to eliminate hazards and to control the hazards with limited consequences.

The current SAPs do not specify BSL or BSO values for either CDF or FDF. However a ‘Plant Damage Frequency’ was defined in earlier SAPs which was taken as being synonymously to CDF. For the purpose of UK ABWR design usage, BSL and BSO are set respectively at 10^{-4} /y and 10^{-5} /y for both CDF and FDF. These values are consistent with the IAEA numerical safety goals for CDFs set for operating plants and new plants [Ref-25.126].

The summed CDF and FDF for the GDA UK ABWR is 4.3E-06 /y.

This level of risk is substantially below the UK ABWR adopted Basic Safety Objective (BSO) value for CDF and FDF.

The combined Level 1 PSA results summarised in this section are based on the integration of the Level 1 at Power, Shutdown, SFP, Internal Fire, Internal Flood and Seismic Hazard analyses. Although additional, simplified assessments have been performed to determine the risk of core/fuel damage from the Fuel Route, non-Reactor Faults, Tornado Missiles, Turbine Missiles and accidental aircraft impact, these results have not been included in the integration as they were found to be very small contributors to the overall core/fuel damage risk.

It is unlikely that future updates of the existing Level 1 PSAs or additional hazards assessments would result in an increase in CDF or FDF such that they would exceed the UK ABWR adopted design targets.

25.15.2.1 Core/Fuel Damage Frequencies

Table 25.15.2-1 shows the core damage frequency of the individual performed PSAs; the associated large release frequency and large early release frequency are also presented for comparison.

Figure 25.15.2-1 and 25.15.2-2 and 25.15.2-3 show the contribution of each fault grouping to overall CDF, FDF or LERF respectively.

Table 25.15.2-1 Risk Measures for UK ABWR PSA

PSA		CDF (/y)	LRF (/y)	LERF (/y)
Internal Events	At Power	2.3E-07	4.6E-08	4.4E-08
	Shutdown	8.7E-08	6.9E-08	5.2E-09
	SFP	4.2E-07	4.8E-08	8.9E-11
	Non-Reactor Fault (including Fuel Route)	n/a	n/a	n/a
Internal Hazard	Fire At Power	5.0E-07	2.7E-07	2.7E-07
	Flooding At Power	1.8E-06	1.8E-07	1.7E-07
External Hazard	Seismic At Power	7.3E-07	6.1E-07	6.0E-07
	Seismic Shutdown	4.2E-08	not calculated	not calculated
	Seismic SFP	4.5E-07	3.9E-07	1.3E-08
	Tornado Missile	5.2E-10	2.4E-10	not calculated
	Turbine Missile	7.1E-10	8.1E-11	not calculated
	Accidental Aircraft Impact	7.9E-10	4.6E-10	not calculated
Total		4.3E-06	1.6E-06	1.1E-06

The risk of core damage from fire, flooding and seismic hazards are shown to dominate the current risk results, with contributions of 12 percent, 42 percent and 29 percent of the overall CDF, respectively.

The above results are based on the initial PSA assessment of each PSA fault group as noted in the Internal Hazards section (Section 25.10) and the External Hazards section (Section 25.11) the preliminary nature of the UK ABWR design with its lack of detailed design information, particularly with regards to the layout of cabling, etc. has resulted in the hazards assessments being based on more conservative assumptions than those in the internal events assessment.

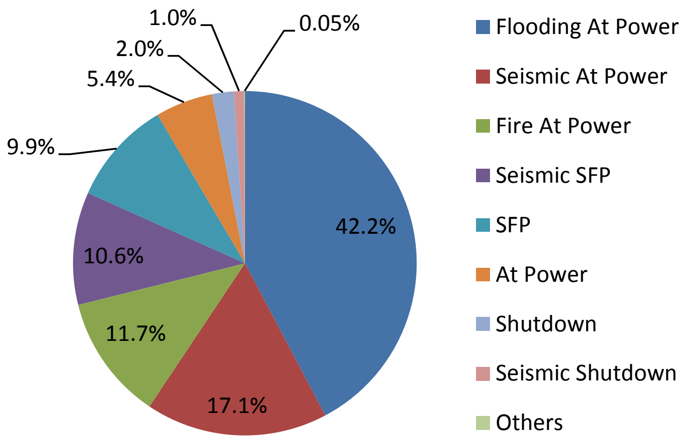


Figure 25.15.2-1 Risk Contribution (CDF) of each PSA

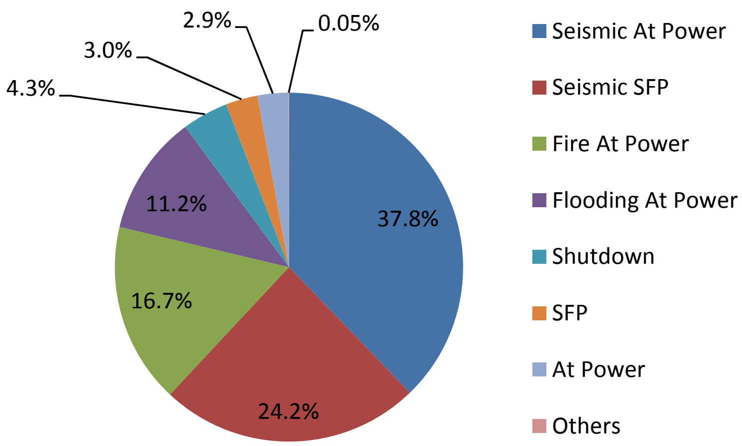


Figure 25.15.2-2 Risk Contribution (LRF) of each PSA

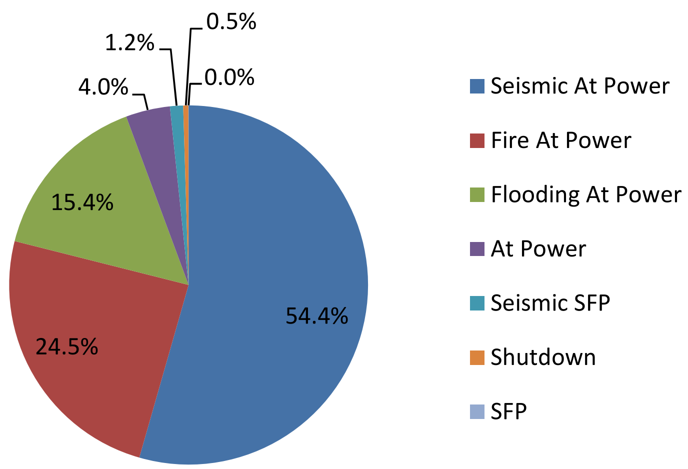


Figure 25.15.2-3 Risk Contribution (LERF) of each PSA

As the design progresses at the post GDA phase, more comprehensive and detailed design information will become available such that the conservative assumptions of the GDA PSAs can be replaced with more realistic assumptions. This will result in less conservative and more realistic results.

As part of the GDA assessment, a review potential plant improvements has been undertaken [Ref-25.11]. These potential plant improvements were used to refine the PSA to provide an estimated measure of plant risk including Societal Risk, and are reflected in the current PSA results.

25.15.2.2 CDF/FDF Distribution

(1) CDF/FDF distribution between events and hazards

Table 25.15.2-2 presents the core damage frequency of main fault groups for at power and shutdown operations and for the Spent Fuel Pool.

Table 25.15.2-2 UK ABWR CDF for Various Events and Hazards

	At power	Shutdown	SFP
Internal events	2.3E-07	8.7E-08	4.2E-07
Internal Hazards	2.3E-06	Lower than 1E-07*1	Lower than 1E-07*1
External Hazards	7.3E-07	4.2E-08	4.5E-07
Total CDF (/y)	**3.3E-06**	**1.3E-07***2	**8.7E-07***2

*1: Scoping analyses are performed for shutdown/SFP fire/flooding PSA as described in Section 25.10.
*2: Total CDF is calculated by the summation of internal events and external hazards.

Internal hazards currently contribute nearly 54 percent to the risk of plant damage, with external hazards contributing a further 29 percent. Internal events currently only contribute 17 percent to the risk of plant damage. As noted in previously, the lack of certain detailed design information has resulted in the hazards assessments being based on more conservative assumptions to those of the internal events. As the design progresses at the post GDA phase and the conservative hazards assessments are replaced with more realistic results, the division of plant damage risk between internal events and internal will be more evenly balanced.

The calculated level of risk of core and fuel damage for all events and hazards groupings are substantially below the UK ABWR adopted Basic Safety Objective (BSO) value for the overall CDF and FDF.

(2) CDF/FDF Risk distribution between at power and shutdown

The calculated level of risk of core and fuel damage during at power is 3,3E-06 /y which represents 77 percent of the overall risk of core or fuel damage. The Spent Fuel Pool represents a further 20 percent (8.7E-07 /y). The risk of core or fuel damage from Shutdown operations currently represents only 3 percent of the overall risk of core or fuel damage. However, at power contribution will be small as the design progresses at the post GDA phase and account is taken of Internal Hazards during Shutdown.

25.15.2.3 Importance Ranking

The GDA UK ABWR PSA hazards assessments by necessity of the design development are currently based upon conservative assumptions, which will be improved as the design evolves. Integration of the Hazards PSA results with those of the Internal Events PSA at this stage is therefore of limited value. However, an initial model integration has been performed on the importance measures to provide additional insights into the robust and adequacy of the UK ABWR design.

The integration was performed by combining the CDF and LRF cutsets and importance measures of the top seven fault group contributors (Internal events at Power, Internal events Shutdown, Internal events SFP, Seismic Hazard at Power, Seismic Hazard SFP, Internal Hazard Fire, Internal Hazard Flooding) in the following process.

(1) Create integrated database

The database for each of above PSA is merged into one database file.

(2) Create integrated cutset

The cutset for each of the PSA is merged into one cutset file. The cutset for each of the PSA is re-truncated and compressed on truncation level of 1.0E-12. The PSA by PSA cutset files could not be merged into single cutset file, which was necessary for importance analysis, when the truncation limit was 1.0E-13 or smaller.

(3) Importance analysis and review the results

Importance analysis is performed by using of CAFTA based on integrated database and cutset generated in the above process. The result of importance analysis is reviewed in accordance with the identification process of risk contributors described in Section 4.1.1 1) b of [Ref-25.11].

In terms of F-V, the following types of basic events are excluded:

- Duration factor events (e.g., plant availability, time fraction of plant outage, POS factors),
- Marker events (sequence markers, accident class markers, release category markers).

In terms of RAW, the following types of basic events are excluded:

- Initiating events: Low probability initiating events tend to have high RAWs. Hitachi-GE understands, however, that the potential risk increase given an initiating event frequency of 1.0 /y is not a meaningful risk insight.
- Component failures contributing to support system initiators or LOCAs: the RAW of such a basic event means a risk increase when the associated initiating event frequency is [365 times fraction of POS duration in a year] /y, and thus tends to be significant. Note that these basic events were also excluded from the system level RAW evaluations in the internal events PSAs [Ref-25.9][Ref-25.75][Ref-25.90].
- CCF events: CCF events tend to have high RAWs due to the low probabilities. Hitachi-GE understands, however, that the potential risk increase given a CCF of redundant components with a 1.0 probability is not a meaningful risk insight.
- Test and Maintenance (T&M) unavailability: Potential risk increase given T&M unavailability of 1.0 is not a meaningful risk insight.

The processes 1 to 3 are performed for both Level 1 and Level 2 PSA, i.e., CDF and LRF. All the basic events with F-V greater than 5 percent and/or RAW greater than 2.0 were screened per the above process.

Although these summed importance measures are likely to change along with a progress of PSA model is developed, consideration of these initially summed importance measures does provide additional insights at this point in the process.

For the purpose of determining a system or component's contribution to safety, two importance measures are commonly used: Fussell-Vesely (F-V): a measure of a component's risk importance, and Risk Achievement Worth (RAW): a measure of a component's safety importance. F-V importance represents the direct effect of the component's unavailability in contributing to core/fuel damage or early release, whilst RAW importance does not represent the component itself, but the defence of the rest of the plant against the failure of component. A high F-V importance indicates that risk can be reduced by improving the component's reliability. A high RAW importance indicates a potential over reliance on a component to provide safety and that additional defence in depth should be considered.

Fussell-Vesely Importance

Table 25.15.2-3 shows the risk importance (F-V) obtained from the integrated cutsets for CDF. Manual initiation of FLSS and unavailability of FLSR are the second and third most important events based on F-V, respectively, and significantly contribute to the Shutdown and SFP risk. The design options associated with these events that were considered important to achieving risk reduction across multiple fault groups.

Loss of a Class 1 AC bus (IE) due to human error is the fourth most important event based on F-V, and significantly contributes to the Shutdown and SFP risk. The design options associated with this event that were considered important to achieving risk reduction across multiple fault groups.

The fifth most important event based on F-V relates to Seismic risk and includes failure of the ceramic insulator. The basic event for this component is significant for full power and the SFP risk. This is evaluated in Table 4.7.5-1 of [Ref-25.11], and no new insights are gained from this review since the failures are specifically related to seismic sequences.

All other events have a F-V of less than 5 percent of the combined (estimated) CDF risk.

Risk Achievement Worth

Table 25.15.2-4 shows the risk importance (RAW) obtained from the integrated cutsets for CDF. The first and second most important events (that are not CCF or initiating events) based on RAW are failure of FLSS RPV injection line check valves E71-F024 and E71-F025 to open. These events impact RPV injection from FLSS and FLSR, and failure of these check valves to open shows up as risk significant in the Shutdown PSA, Fire PSA, and Flood PSA. This is evaluated for each of these hazards in Table 4.3.5-1, Table 4.5.5-1, and Table 4.6.5-1 of [Ref-25.11], respectively. No new insights are gathered from this review since alternate RPV injection lines are available given check valve failure and the F-V importance for these events is low.

The third most important event for RAW is plugging of FLSS injection line manual valve E71-F026. This event impacts RPV injection from FLSS and FLSR, and plugging of this manual valve shows up as risk significant in the Shutdown PSA, Fire PSA, and Flood PSA. This is evaluated for each of these hazards in Table 4.3.5-1, 4.5.5-1, and 4.6.5-1 of [Ref-25.11], respectively. No new insights are gathered from this review since alternate RPV injection lines are available given manual valve failure and the F-V importance for this event is low.

The fourth most important event for RAW is plugging of FLSS orifice E71-D004. This event impacts RPV injection from FLSS and FLSR, and plugging of this orifice shows up as risk significant in the Fire PSA and Flood PSA. This is evaluated for each of these hazards in Table 4.5.5-1 and Table 4.6.5-1 of [Ref-25.11], respectively. No new insights are gathered from this review since alternate RPV injection lines are available given orifice failure and the F-V importance for this event is low.

The fifth and sixth most important events for RAW are failure of FLSS RPV injection line check valve E71-F023 to open and plugging of FLSS RPV injection line manual valve F022, respectively. These events impact RPV injection from FLSS and FLSR, and failure of these valves shows up as risk significant in the

Fire PSA and Flood PSA. This is evaluated for each of these hazards in Table 4.5.5-1 and Table 4.6.5-1 of [Ref-25.11], respectively. No new insights are gathered from this review since alternate RPV injection lines are available given check valve failure and the F-V importance for these events is low.

The seventh most important event for RAW is plugging of FLSS/FLSR SFP spray sparger E71-D009. This event impacts SFP injection from FLSS and FLSR, and failure of this sparger shows up as risk significant in the Shutdown PSA and SFP PSA. This is evaluated for each of these hazards in Table 4.3.5-1 and Table 4.4.5-1 of [Ref-25.11], respectively. No new insights are gathered from this review since alternate SFP injection lines are available given check valve failure and the F-V importance for these events is low.

The eighth and ninth most important events for RAW are failure of FLSS MOV E71-F021B to open and close. These events impact RPV injection from FLSS and FLSR, and failure of these valves shows up as risk significant in the Flood PSA. This is evaluated in Table 4.6.5-1 of [Ref-25.11], and no new insights are gathered from this review since the failures are specifically related to flood sequences.

The tenth and eleventh most important events for RAW are failure of FLSS MOV E71-F021A to open and close. These events impact RPV injection from FLSS and FLSR, and failure of these valves shows up as risk significant in the Flood PSA. This is evaluated in Table 4.6.5-1 of [Ref-25.11], and no new insights are gathered from this review since the failures are specifically related to flood sequences.

Table 25.15.2-3 Risk Importance From Integrated Cutsets for CDF (F-V > 0.05)

Event	F-V	Description
B21-SRV-CC-27-F001BEGKMSU	0.131	CCF of 2/7 Safety Relief Valve F001BEGKMSU Fail to Open
___-HFE-FC-FL	0.105	Failure of manual initiation of FLSS
FLSR-SD	0.099	FLSR (Mobile Injection Facility) Unavailability
___-HFE-SB-AC_A	0.055	Loss of a Class 1 AC bus
SF-___-CIN-C-G08	0.053	SEISMIC FRAGILITY FOR %G08: Ceramic Insulator

Table 25.15.2-4 Risk Importance From Integrated Cutsets for CDF (Top 10 RAW for Non-screened Events)

Event	RAW	Description
E71-CV_-CC-__-F024	349.8	Check Valve F024 Fail to Open
E71-CV_-CC-__-F025	349.8	Check Valve F025 Fail to Open
E71-MV_-PG-__-F026	320.0	Manual Valve F026 Plugged
E71-OF_-PG-__-D004	198.7	Orifice D004 Plugged
E71-CV_-CC-__-F023	189.5	Check Valve F023 Fail to Open
E71-MV_-PG-__-F022	169.5	Manual Valve F022 Plugged
E71-SR_-PG-__-D009	155.4	Strainer D009 Plugged
E71-MOV-CC-__-F021B	7.0	Motor-Operated Valve F021B Fail to Open
E71-MOV-OO-__-F021B	7.0	Motor-Operated Valve F021B Fail to Close
E71-MOV-CC-__-F021A	7.0	Motor-Operated Valve F021A Fail to Open
E71-MOV-OO-__-F021A	7.0	Motor-Operated Valve F021A Fail to Close

25.15.3 Summary of Level 2 PSA Results

25.15.3.1 Large Release Frequency

As shown in the Table 25.15.2-1 and Figure 25.15.2-2, the risk of Large Release from fire, flooding and seismic hazards are shown to dominate with contributions 17 percent, 11 percent and 62 percent of the overall LRF, respectively.

The above results are based on the initial PSA assessment of each PSA fault group as noted in the Internal Hazards section (Section 25.10) and the External Hazards section (Section 25.11) the preliminary nature of the UK ABWR design with its lack of detailed design information, particularly with regards to the layout of cabling, etc. has resulted in the hazards assessments being based on more conservative assumptions to those the internal events.

As the design progresses at the post GDA phase more comprehensive and detailed design information will become available such that the conservative assumptions of the GDA PSAs can be replaced with more realistic assumptions, this will result in less conservative and more realistic results.

As part of the GDA assessment, and in lieu of detailed design information, a review identifying potential plant improvements has been undertaken [Ref-25.11]. These potential plant improvements were used to refine the PSA to provide an estimated measure of plant risk including Societal Risk, and are reflected in the current PSA results.

25.15.3.2 LRF Distribution

(1) LRF distribution between events and hazards

Table 25.15.3-1 presents the core damage frequency of main fault groups for at power and shutdown operations and for the Spent Fuel Pool.

Table 25.15.3-1 UK ABWR LRF for Various Events and Hazards

	At power	Shutdown	SFP
Internal events	4.6E-08	6.9E-08	4.8E-08
Internal Hazards	4.5E-07	Lower than 1E-07*1	Lower than 1E-07*1
External Hazards	6.1E-07	4.2E-08*2	3.9E-07
Total LRF (/y)	**1.1E-06**	**1.1E-07*3**	**4.4E-07*3**

*1: Scoping analyses are performed for shutdown/SFP fire/flooding PSA as described in Section 25.10.
*2: LRF for shutdown external hazards PSA is assume to be equivalent to FDF.
*3: Total LRF is calculated by the summation of internal events and external hazards.

Internal hazards currently contribute nearly 28 percent to the risk of large release, with external hazards a further 62 percent. Internal events currently only contribute 10 percent to the risk of large release. As noted in previously, the lack of certain detailed design information has resulted in the hazards assessments being based on more conservative assumptions to those of the internal events. As the design progresses at the post GDA phase and the conservative hazards assessments are replaced with more realistic results, the division of plant damage risk between internal events and internal will be more evenly balanced.

The calculated level of risk of release for all events and hazards groupings are substantially below the UK ABWR adopted Basic Safety Objective (BSO) value for the overall LRF.

(2) LRF Risk distribution between at power and shutdown

The calculated level of risk of large release during at power is 1.1E-6 /y which represents 69 percent of the overall risk of large release. The Spent Fuel Pool represents a further 28 percent (4.4E-07 /y). The risk of large release from Shutdown operations currently represents 3 percent (1.1E-07 /y) of the overall risk of large release, however the shutdown contribution will rise as the design progresses at the post GDA phase and account is taken of Internal Hazards during Shutdown.

25.15.3.3 Importance ranking

Table 25.15.3-2 shows the risk importance (F-V) obtained from the integrated cutsets for LRF. F-V Importance Measures for LRF include similar basic events as the CDF discussion in Section 25.15.2.3, including the Ceramic Insulator as a representative component for loss of all offsite power, FLSS tank and CCF for SRVs. Other new events in LRF involve the probability that a small LOCA occurs in the drain and instrumentation tap or RVI piping following a Seismic Event, and significantly contribute to both full power and SFP PSAs. These items are discussed in Table 4.7.5-1 of [Ref-25.11], and no new insights are gained from this review since the failures are specifically related to seismic sequences.

Additional new events in LRF relate to Seismic risk and involve failure of the FLSS tank and CUW piping. The basic events for these components are significant for full power and the SFP risk. This is evaluated in Table 4.7.5-1 of [Ref-25.11], and no new insights are gained from this review since the failures are specifically related to seismic sequences. All other events have a F-V of less than 5 percent of the combined (estimated) LRF risk.

RAW importance measures for LRF include similar basic events as the CDF discussion in Section 25.15.2.3, including failure to open FLSS RPV injection line check valves E71-F024, F025 and plugging of FLSS RPV injection line manual valve E71-F026, FLSS/FLSR spray sparger E71-D009, and FLSS orifice E71-D004. Failure to open FLSS RPV injection line check valve E71-F023, plugging of FLSS RPV manual valve E71-F022, and failure to open/close FLSS motor-operated valves E71-F021A/B do not contribute to LRF.

New events for LRF involve the loss of function of DC power buses DC-B/B1 and DC-B/B2. Failure of these buses impacts RPV/PCV injection from FLSS. As discussed in Table 4.5.5-1 of [Ref-25.11], failure of these buses was significant based on the unrefined internal events at power Fire PSA; however, this insight is not applicable to the refined Fire PSA results. On the other hand, failure of these buses shows up as significant in the Flood PSA. These items are discussed in Table 4.6.5-1 of [Ref-25.11], and no new insights are gained from this review since the failures are specifically related to flood sequences.

Additional new events for LRF involve the failure of the Class 1 control panel, instrumentation rack, FLSS/FLSR piping following a Seismic event. These items are discussed in Table 4.7.5-1 of [Ref-25.11], and no new insights are gained from this review since the failures are specifically related to seismic sequences.

Table 25.15.3-2 Risk Importance from Integrated Cutsets for LRF (F-V > 0.05)

Event	F-V	Description
WEIGHT-CUW-B_O	0.424	Weight Factor CUW BOC Outside of Outboard Isolation Valve
WEIGHT-LOCA-S2	0.256	Weight Factor Seismic Induced Small LOCA Drain and Instrumentation Tap
WEIGHT-LOCA-S1-BDL	0.112	Weight Factor Medium LOCA for BDL
SF-G31-PP-C-G09	0.101	SEISMIC FRAGILITY FOR %G09: Class 3 CUW Piping
SF-___-CIN-C-G08	0.094	SEISMIC FRAGILITY FOR %G08: Ceramic Insulator
SF-G31-PP-C-G10	0.080	SEISMIC FRAGILITY FOR %G10: Class 3 CUW Piping
SF-G31-PP-C-G08	0.076	SEISMIC FRAGILITY FOR %G08: Class 3 CUW Piping
WEIGHT-LOCA-S2-RVID	0.074	Weight Factor Seismic Induced Small LOCA RVI div.D
WEIGHT-LOCA-S2-RVIC	0.071	Weight Factor Seismic Induced Small LOCA RVI div.C
SF-___-CIN-C-G09	0.070	SEISMIC FRAGILITY FOR %G09: Ceramic Insulator
SF-E71-TKU-C-G08	0.064	SEISMIC FRAGILITY FOR %G08: FLSS Tank (Unpressurized)
WEIGHT-LOCA-S2-RVIA	0.060	Weight Factor Seismic Induced Small LOCA RVI div.A
B21-SRV-CC-27-F001BEGKMSU	0.057	CCF of 2/7 Safety Relief Valve F001BEGKMSU Fail to Open
WEIGHT-LOCA-S2-RVIB	0.055	Weight Factor Seismic Induced Small LOCA RVI div.B
HBURN_BF	0.054	Hydrogen burning*
SF-G31-PP-C-G11	0.052	SEISMIC FRAGILITY FOR %G11: Class 3 CUW Piping

*: This event is considered to map the relevant source term in SFP Level 2 PSA. The end states correnpond to the fission product large release regardless of the occurrence of hydrogen burining. Therefore, ALARP discussion on this is not provided.

Table 25.15.3-3 Risk Importance from Integrated Cutsets for LRF (Top 10 RAW for Non-screened Events)

Event	RAW	Description
E71-CV_-CC-__-F024	382.8	Check Valve F024 Fail to Open
E71-CV_-CC-__-F025	382.8	Check Valve F025 Fail to Open
E71-MV_-PG-__-F026	362.8	Manual Valve F026 Plugged
E71-SR_-PG-__-D009	304.1	Strainer D009 Plugged
E71-OF_-PG-__-D004	15.4	Orifice D004 Plugged
R42-BSD-LF-__-DC-B/B2	11.0	Bus (DC Power) DC-B/B2 Loss of Function
R42-BSD-LF-__-DC-B/B1	9.3	Bus (DC Power) DC-B/B1 Loss of Function
SF-___-CP-C1-C-G04	8.2	SEISMIC FRAGILITY FOR %G04: Class 1 Control Panel
SF-___-IR-C1-C-G04	8.2	SEISMIC FRAGILITY FOR %G04: Class 1 Instrumentation Rack
E71-CV_-CC-__-F023	6.9	Check Valve F023 Fail to Open
SF-E71-PP-C-G04	6.8	SEISMIC FRAGILITY FOR %G04: FLSS Piping
SF-E72-PP-C-G04	6.8	SEISMIC FRAGILITY FOR %G04: FLSR Piping

25.15.4 Summary of Level 3 PSA Results (Offsite Consequence Assessment)

The objectives of the Level 3 PSA for the UK ABWR are to:

- Demonstrate compliance against UK numerical risk targets for fault analysis. These numerical targets are defined by the BSLs and BSOs for NSEDPs [Ref-25.58] equivalent to Target 7, Target 8, and Target 9 of the SAPs.
- Present the plant risk profile based on the Level 3 PSA consequences calculations and the Level 2 PSA release category frequencies.

Previous sections of this report have presented the contributions from each 'fault group' individually. The compiled risk profile considers all the contributions.

Comparisons with the numerical targets are considered for each event group separately and then the combined risk is compiled and assessed against the numerical targets. The events that contribute significantly to each event group and to the compiled plant risk profile are presented for each of the numerical targets.

The event groups explicitly considered in the plant risk profile for the Level 3 PSA are:

(1) IE at Power leading to fuel melt,

(2) IE during Shutdown leading to fuel melt,

(3) SFP IE leading to fuel melt,

(4) Events not leading to fuel melt, consisting of:

- PSA 'success' sequence groups, i.e. those PSA sequences that do not lead to fuel melt,
- SFP and Fuel Route fault groups from the DBA and BDBA that do not lead to fuel melt,
- Non-Reactor fault groups from the DBA and BDBA.

(5) Internal hazard events leading to fuel melt, consisting of:

- Fire as an internal hazard initiator leading to fuel melt for the reactor at Power,
- Flood as an internal hazard initiator leading to fuel melt for the reactor at Power.

(6) External hazard events leading to fuel melt, consisting of:

- Seismic event as an external hazard initiator leading to fuel melt for the reactor at Power, and
- Seismic event as an external hazard initiator leading to fuel melt in the SFP (and any consequential impact on the operating reactor).

Overall, the frequency of fuel melt resulting from Internal Hazards (Fire and Flood) and External Hazards (seismic event) affecting the reactor at Power is a factor 10 higher than that from all IE at Power. This is due to the conservative nature of the Internal Hazard and External Hazard PSAs in the GDA.

The following fault groups are not explicitly considered in the compiled risk profile of the Level 3 PSA, as the Level 2 PSA is performed semi-quantitatively (or it is qualitatively argued risk is insignificant) and does not lead to explicit release category frequencies:

- Fire as an internal hazard initiator leading to fuel melt for the shutdown reactor and SFP,
- Flood as an internal hazard initiator leading to fuel melt for the shutdown reactor and SFP,
- Seismic event as an external hazard initiator leading to fuel melt for the shutdown reactor,
- Fire as an internal hazard initiator leading to fuel melt for the SFP, and

- Flood as an internal hazard initiator leading to fuel melt for the SFP.

25.15.4.1 Facility Dose Bands (Target 8)

The assessment against the facility dose band targets is presented in Table 25.15.4-1 and shown graphically in Figure 25.15.4-1. The contributions of each event group to the total are shown graphically in Figure 25.15.4-2 to Figure 25.15.4-9. The individual contributors to the total frequency in each dose band are given in Table 25.15.4-2 to Table 25.15.4-6.

The assessment against the facility dose band targets is summarised as follows:

- The frequency allocated to the 0.1 to 1 mSv dose band is 1.39E-03 /y. This is 13.9 percent of the BSO and 0.139 percent of the BSL.
- The frequency allocated to the 1 to 10 mSv dose band is 2.43E-04 /y, which is 24.3 percent of the BSO and 0.243 percent of the BSL. This frequency is dominated by the events not leading to fuel melt. The dominant contributors, see Table 25.15.4-3, are:
    - R* : Success sequence by RHR operation (containment intact) with fuel perforation {1.78E-04 /y}, and
    - FH : Fuel Handling Accident with SGTS unsuccessful {6.51E-05 /y}.

  No other contributors are >1 percent of the BSO.
- The frequency allocated to the 10 to 100 mSv dose band is 6.15E-06 /y, which is less than 6.2 percent of the BSO and 0.062 percent of the BSL. This frequency is dominated by the events not leading to fuel melt. The dominant contributors, see Table 25.15.4-4, are:
    - CDPF : Cask drop onto cask pit or preparation pit with SGTS unsuccessful {7.6E-07 /y}, and
    - C* : Success sequence by containment venting (containment intact) with fuel perforation {5.39E-06 /y}.

  No other contributors are >1 percent of the BSO.
- The frequency allocated to the 100 to 1,000 mSv dose band is 1.52E-06 /y, which is 15.2 percent of the BSO and 0.152 percent of the BSL. The dominant contributors, see Table 25.15.4-5, are:
    - IF3 Filtered Containment Venting (TQUV) {1.67E-07 /y},
    - IFL3 Filtered Containment Venting (TQUV) {1.16E-07 /y},
    - P3 Filtered Containment Venting (TQUV no DW sprays) {1.31E-07 /y}, and
    - ES 3 Filtered Containment Venting (no DW sprays) {1.31E-07 /y}.

  No other contributors are >4 percent of the BSO.
- The frequency assigned to the highest dose band (>1,000 mSv) is 2.27E-06 /y or 227 percent of the BSO and 2.27 percent of the BSL. This frequency is dominated by the external hazard which, together, contribute 1.04E-06 /y or 104 percent of the BSO. The remainder contribute 1.23E-06 or 123 percent of the BSO and are dominated by the PSA ‘success sequence’ groups with fuel perforation. The dominant contributors, see Table 25.15.4-6, are:
    - IFL15 Direct Debris Interaction (TQUX) {2.36E-07 /y},
    - ES 13 Containment Bypass (S3E) {2.32E-07 /y},
    - SBO-SFP02 {2.22E-07 /y},
    - ES 12 RPV rupture (S4) {1.43E-07 /y},

- IF14 S/P Bypass (TNQUV) {1.24E-07 /y},
- CP*: Success sequence with containment overpressure failure (with success of VSS) with fuel perforation {1.21E-07 /y},
- ES10-2 PCV Isolation Failure (AE) {8.97E-08 /y},
- SBO-SFP-LOCA02 {8.10E-08 /y},
- IF13 Containment Bypass (S3E) {7.54E-08 /y},
- BO-SDO-L {6.69E-08 /y}
- IF4 Early Containment Failure (AC) {5.47E-08 /y},
- IF10-2 PCV Isolation Failure (AE) {5.47E-08 /y},
- IFL10-1 PCV Isolation Failure (TQUV) {5.13E-08 /y},
- IFL12 RPV rupture (S4) {4.39E-08 /y},
- IFL6 Late Containment Failure with PCV spray (AE) {4.33E-08 /y}, and
- ES16 Long Term SBO (TB in-vessel FCI) {4.28E-08 /y}.

No other contributors are >5 percent of the BSO.

Table 25.15.4-1 Compiled table for comparison against facility dose bands (Target 8)

Facility dose band (mSv)	Initiating Event Group	Summed Frequency (/y)	Total frequency (/y)	BSO
0.1 to 1	Events not leading to fuel melt	1.39E-03	**1.39E-03**	1.0E-2
1 to 10	IE at Power Level 2 PSA Events not leading to fuel melt Internal Hazards Level 2 PSA (Fire) Internal Hazards Level 2 PSA (Flood) Seismic Event – Reactor at Power	3.84E-08 2.43E-04 2.33E-08 7.71E-09 2.12E-08	**2.43E-04**	1.0E-3
10 to 100	Events not leading to fuel melt	6.15E-06	**6.15E-06**	1.0E-4
100 to 1,000	IE at Power Level 2 PSA Internal Hazards Level 2 PSA (Fire) Internal Hazards Level 2 PSA (Flood) Seismic Event – Reactor at Power	1.31E-07 1.67E-07 1.16E-06 6.14E-08	**1.52E-06**	1.0E-5
> 1,000	IE at Power Level 2 PSA IE at Shutdown Level 2 PSA SFP IE Level 2 PSA Events not leading to fuel melt Internal Hazards Level 2 PSA (Fire) Internal Hazards Level 2 PSA (Flood) Seismic Event – Reactor at Power Seismic Event – Spent Fuel Pool	6.49E-08 7.03E-08 4.79E-08 1.66E-07 3.05E-07 5.78E-07 6.51E-07 3.86E-07	**2.27E-06**	1.0E-6
Summated frequency of release categories /y			**1.64E-03**	

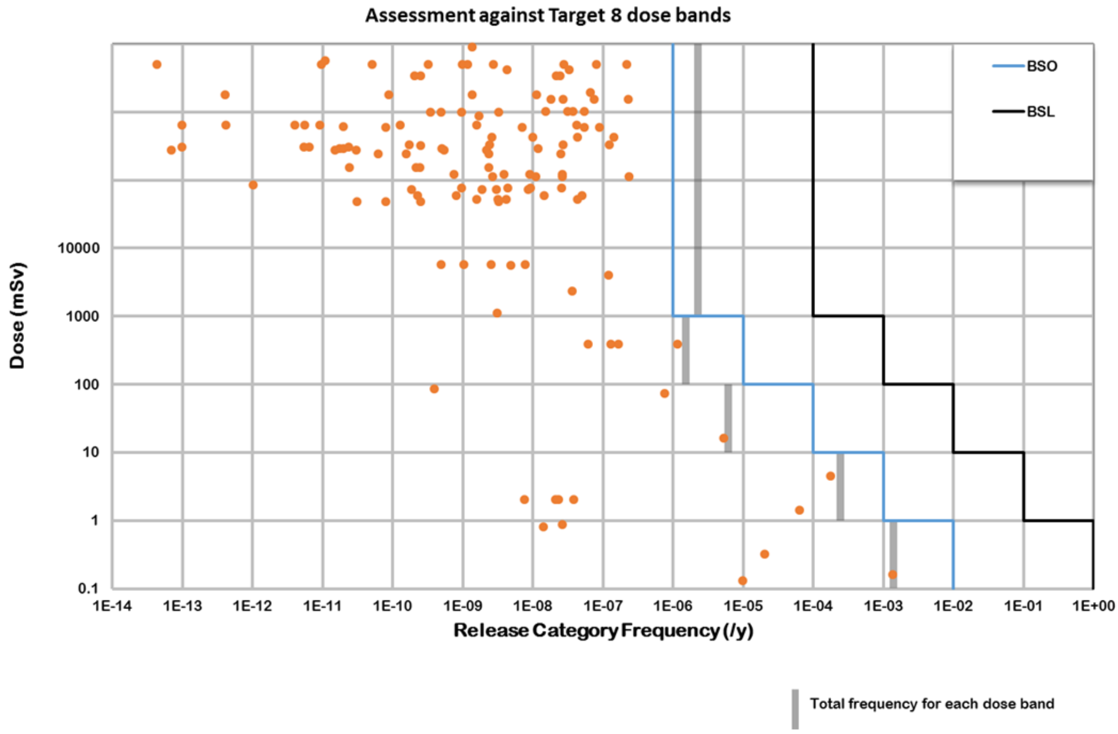


Figure 25.15.4-1 Compiled Assessment against Facility Dose Bands (for Target 8)

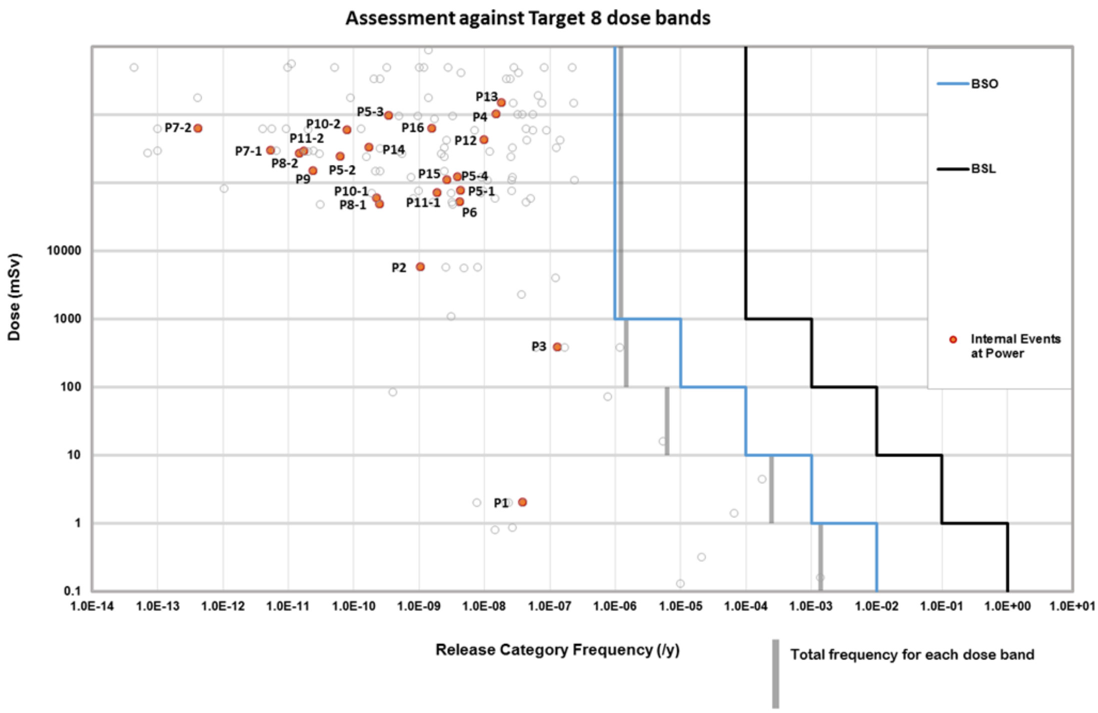


Figure 25.15.4-2 Assessment against Facility Dose Bands (for Target 8) Showing Contribution from IE at Power Leading to Fuel Melt

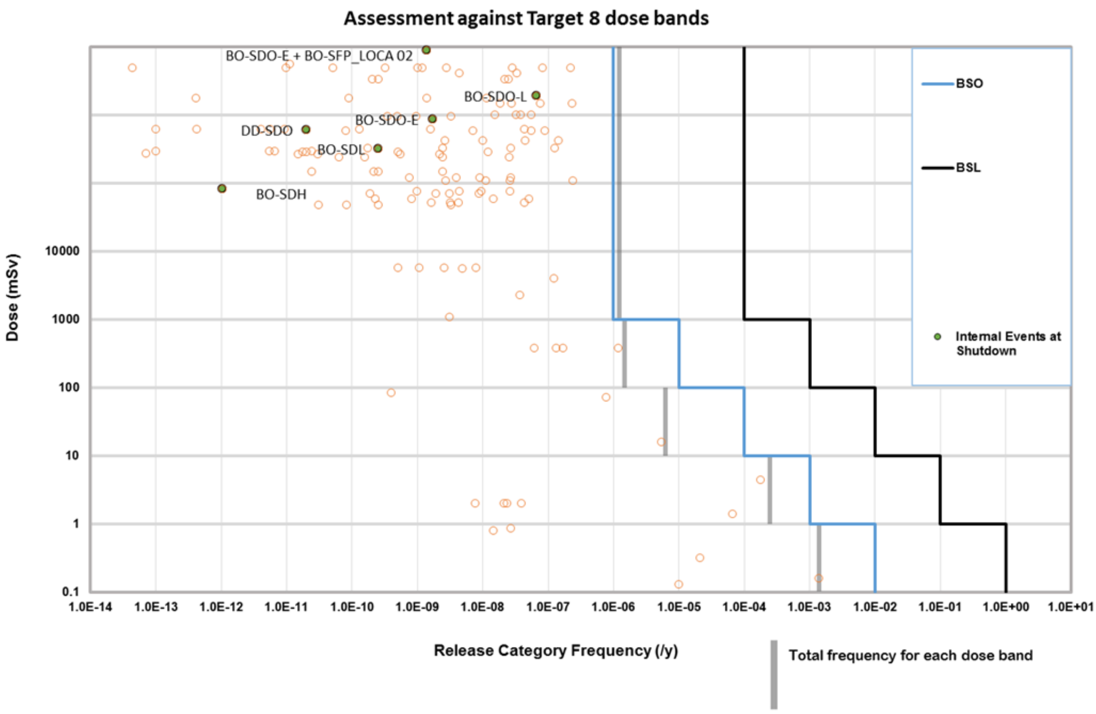


Figure 25.15.4-3 Assessment against Facility Dose Bands (for Target 8) Showing Contribution from IE at Shutdown Leading to Fuel Melt

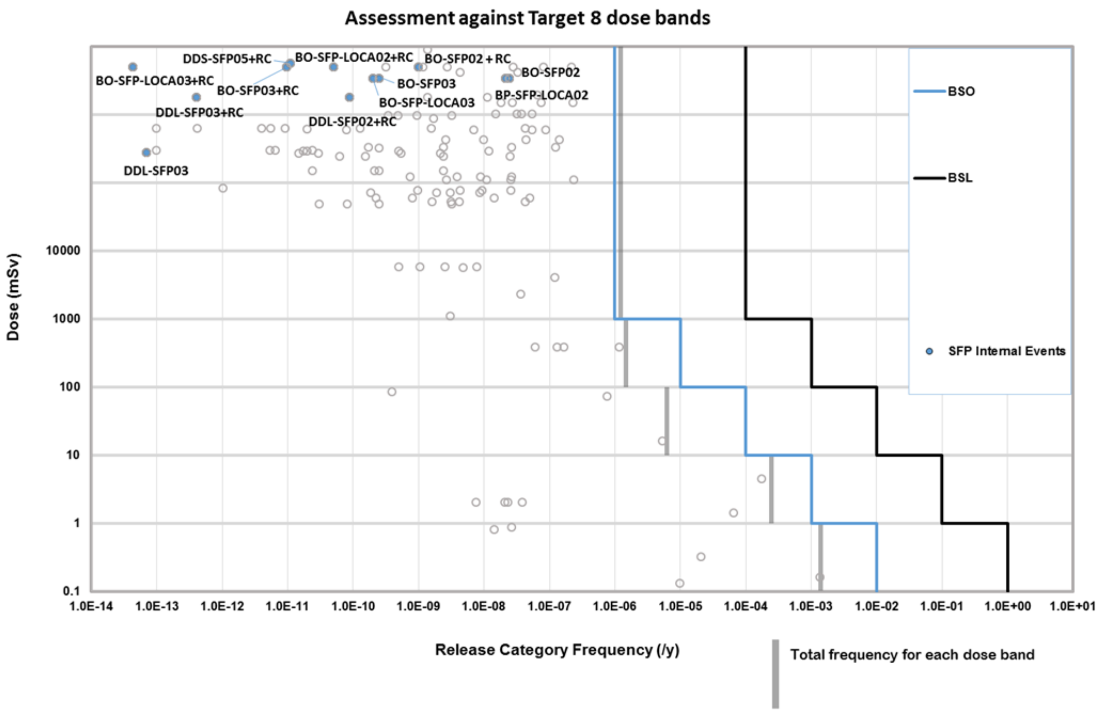


Figure 25.15.4-4 Assessment against Facility Dose Bands (for Target 8) Showing Contribution from SFP IE Leading to Fuel Melt

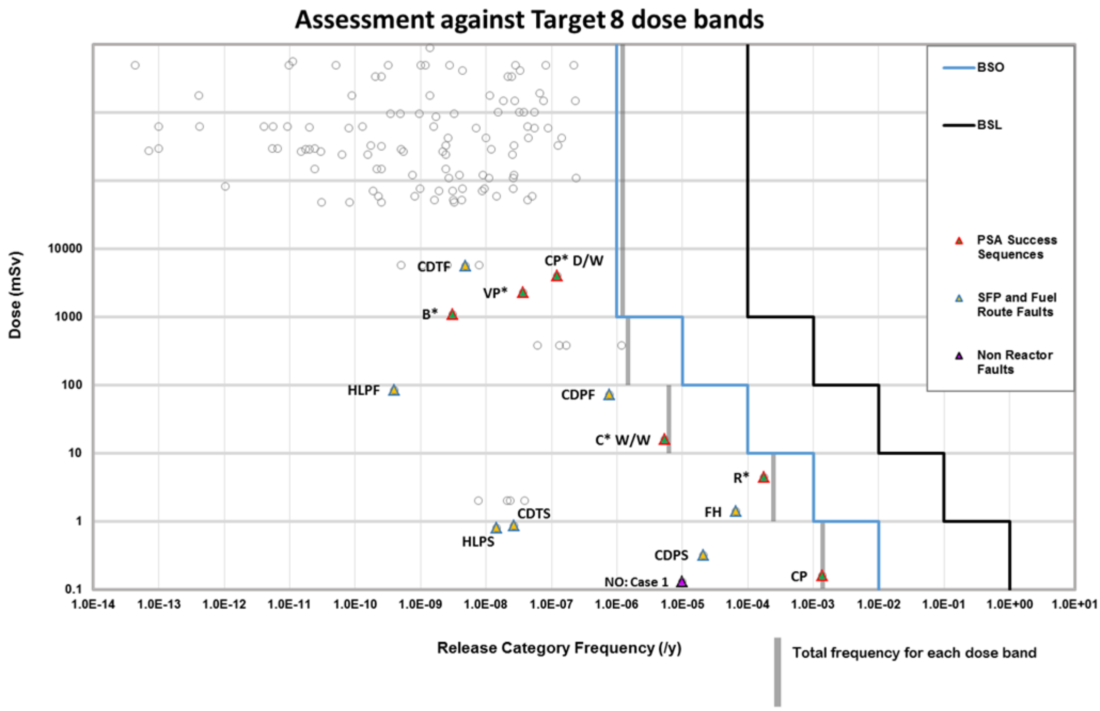


Figure 25.15.4-5 Assessment against Facility Dose Bands (for Target 8) Showing Contribution from Events not Leading to Fuel Melt

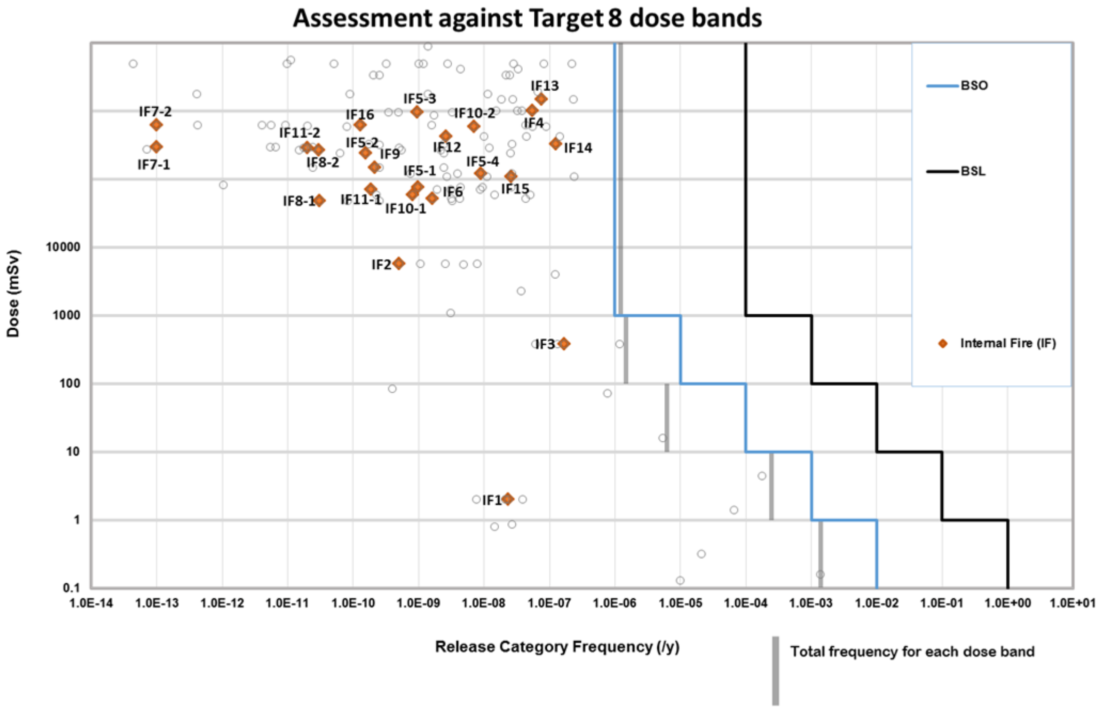


Figure 25.15.4-6 Assessment against Facility Dose Bands (for Target 8) Showing Contribution from Internal Hazards: Fire Leading to Fuel Melt

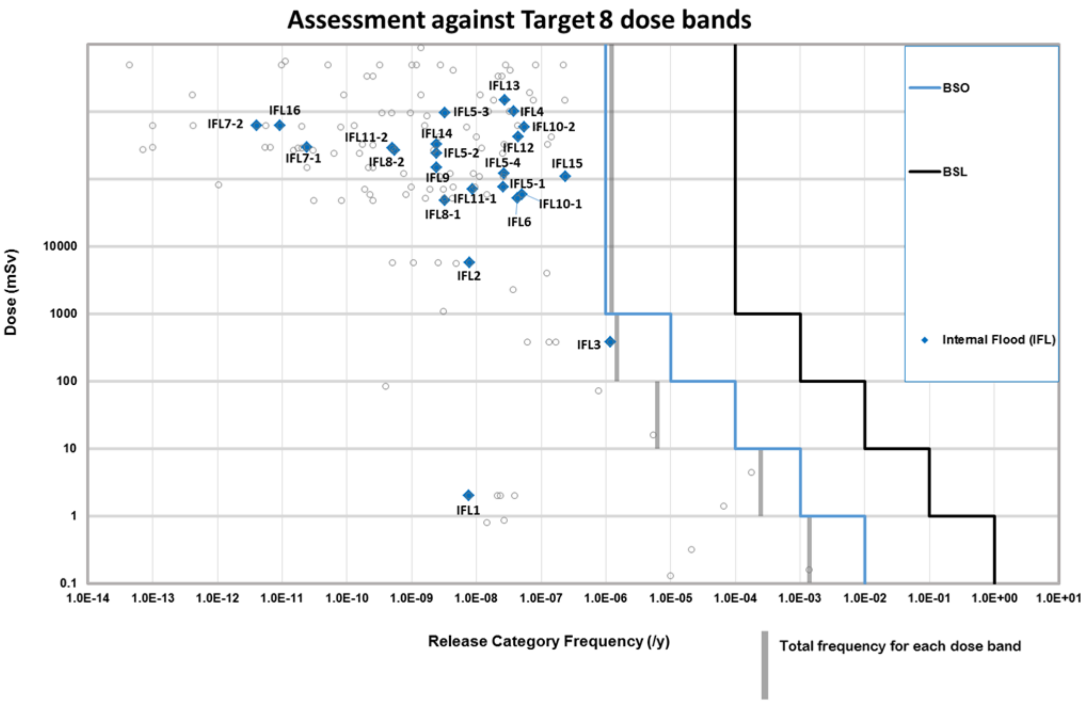


Figure 25.15.4-7 Assessment against Facility Dose Bands (for Target 8) Showing Contribution from Internal Hazards: Flood Leading to Fuel Melt

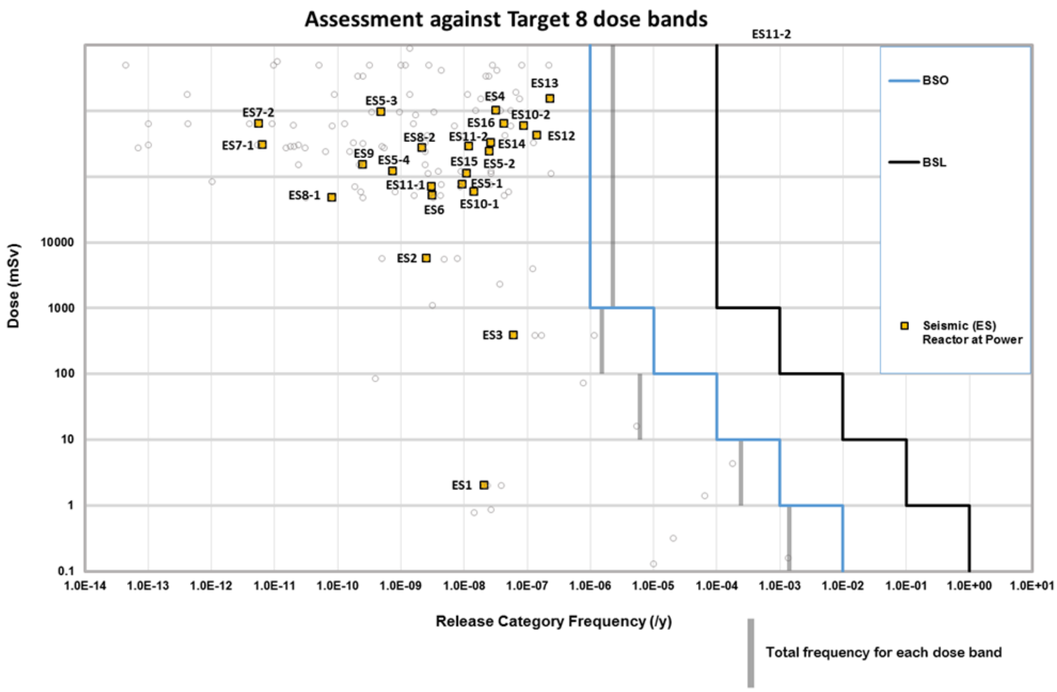


Figure 25.15.4-8 Assessment against Facility Dose Bands (for Target 8) Showing Contribution from External Hazards: Seismic Event Affecting Reactor at Power Leading to Fuel Melt

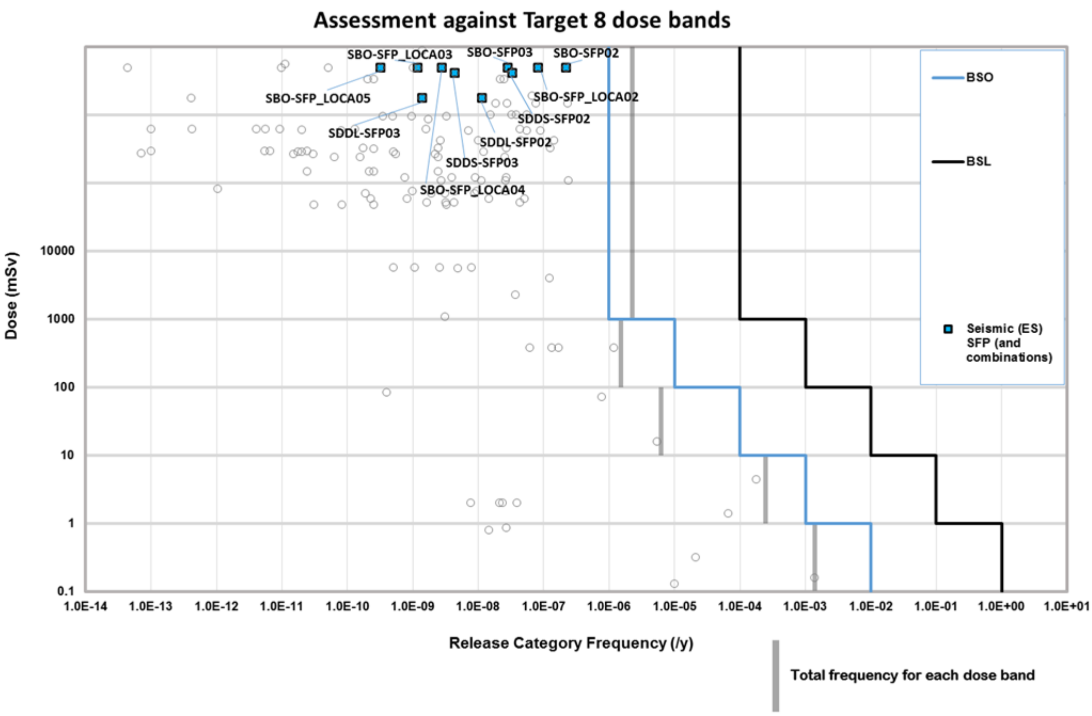


Figure 25.15.4-9 Assessment against Facility Dose Bands (for Target 8) Showing Contribution from External Hazards: Seismic Event Affecting SFP Leading to Fuel Melt

Table 25.15.4-2 Breakdown of Contributors to Facility Dose Band: 0.1 to 1.0 mSv

release category / sequence group / fault group	Level 3 PSA case	Frequency (/y)	% of total band frequency
none			
IE at Power Level 2 PSA Sub-total		0.00E+00	
none			
IE at Shutdown Level 2 PSA Sub-total		0.00E+00	
none			
SFP IE Level 2 PSA Sub-total		0.00E+00	
CP: Success sequence with containment overpressure failure (with success of VSS)	CP	1.36E-03	97.79 %
HLPS : Heavy Load (e.g. cask) drop into spent fuel pool with SGTS successful	HLPS	1.43E-08	< 0.01 %
CDPS : Cask drop onto cask pit or preparation pit with SGTS successful	CDPS	2.07E-05	1.49 %
CDTS : Cask drop onto truck bay shaft or LOOP with SGTS successful	CDTS	2.67E-08	< 0.01 %
NO: unmitigated. Offgas System Failure unmitigated case	NO: Case 2	1.00E-05	0.72 %
Events not leading to fuel melt Sub-total		1.39E-03	
none			
Internal Hazards Level 2 PSAs Sub-total		0.00E+00	
none			
External Hazards Level 2 PSAs Sub-total		0.00E+00	
Total Frequency /y		**1.39E-03**	
Total as % of BSO		**13.9 %**	
BSO		1.00E-02	
BSL		1.00E-00	

Table 25.15.4-3 Breakdown of Contributors to Facility Dose Band: 1 to 10 mSv

release category / sequence group / fault group	Level 3 PSA case	Frequency (/y)	% of total band frequency
1 Containment Leakage from D/W	P1	3.84E-08	0.02 %
IE at Power Level 2 PSA Sub-total		3.84E-08	
none			
IE at Shutdown Level 2 PSA Sub-total		0.00E+00	
none			
SFP IE Level 2 PSA Sub-total		0.00E+00	
R* : Success sequence by RHR operation (containment intact) with fuel perforation	R*	1.78E-04	73.19 %
FH : Fuel Handling Accident with SGTS unsuccessful	FH	6.51E-05	26.77 %
Events not leading to fuel melt Sub-total		2.43E-04	
IF 1 Containment Leakage from D/W (with RPV breach)	P1	2.33E-08	0.01 %
IFL 1 Containment Leakage from D/W (with RPV breach)	P1	7.71E-09	< 0.01 %
Internal Hazards Level 2 PSAs Sub-total		3.10E-08	
ES 1 Containment Leakage from D/W	P1	2.12E-08	0.01 %
External Hazards Level 2 PSAs Sub-total		2.12E-08	
Total Frequency /y		**2.43E-04**	
Total as % of BSO		**24.3 %**	
BSO		1.00E-03	
BSL		1.00E-01	

Table 25.15.4-4 Breakdown of Contributors to Facility Dose Band: 10 to 100 mSv

release category / sequence group / fault group	Level 3 PSA case	Frequency (/y)	% of total band frequency
none			
IE at Power Level 2 PSA Sub-total		0.00E+00	
none			
IE at Shutdown Level 2 PSA Sub-total		0.00E+00	
none			
SFP IE Level 2 PSA Sub-total		0.00E+00	
C* W/W : Success sequence by containment venting (containment intact) with fuel perforation	C*	5.39E-06	87.64 %
HLPF : Heavy Load (e.g. cask) drop into spent fuel pool with SGTS unsuccessful	HLPF	3.93E-10	0.01 %
CDPF : Cask drop onto cask pit or preparation pit with SGTS unsuccessful	CDPF	7.60E-07	12.36 %
Events not leading to fuel melt Sub-total		6.15E-06	
none			
Internal Hazards Level 2 PSAs Sub-total		0.00E+00	
none			
External Hazards Level 2 PSAs Sub-total		0.00E+00	
Total Frequency /y		**6.15E-06**	
Total as % of BSO		**6.2 %**	
BSO		1.00E-04	
BSL		1.00E-02	

Table 25.15.4-5 Breakdown of Contributors to Facility Dose Band: 100 to 1,000 mSv

release category / sequence group / fault group	Level 3 PSA case	Frequency (/y)	% of total band frequency
3 Filtered Containment Venting (no DW sprays)	P3	1.31E-07	8.62 %
IE at Power Level 2 PSA Sub-total		1.31E-07	
none			
IE at Shutdown Level 2 PSA Sub-total		0.00E+00	
none			
SFP IE Level 2 PSA Sub-total		0.00E+00	
none			
Events not leading to fuel melt Sub-total		0.00E+00	
IF 3 Filtered Containment Venting (with RPV breach)	P3	1.67E-07	10.99 %
IFL 3 Filtered Containment Venting (with RPV breach)	P3	1.16E-06	76.35 %
Internal Hazards Level 2 PSAs Sub-total		1.33E-06	
ES 3 Filtered Containment Venting (no DW sprays)	P3	6.14E-08	4.04 %
External Hazards Level 2 PSAs Sub-total		6.14E-08	
Total Frequency /y		**1.52E-06**	
Total as % of BSO		**15.2 %**	
BSO		1.00E-05	
BSL		1.00E-03	

Table 25.15.4-6 Breakdown of Contributors to Facility Dose Band: > 1,000 mSv (1/5)

release category / sequence group / fault group	Level 3 PSA case	Frequency (/y)	% of total band frequency
2 Containment Venting (TQUV no DW sprays)	P2	1.05E-09	0.05 %
4 Early Containment Failure (AC)	P4	1.54E-08	0.68 %
5-1 Late Containment Failure (TQUV)	P5-1	4.39E-09	0.19 %
5-2 Late Containment Failure (AE)	P5-2	6.33E-11	< 0.01 %
5-3 Late Containment Failure (AW-LP)	P5-3	3.47E-10	0.02 %
5-4 late Containment Failure (TW-LP)	P5-4	3.93E-09	0.17 %
6 Late Containment Failure with PCV spray (AE)	P6	4.21E-09	0.19 %
7-1 In-vessel Fuel-Coolant Interaction (TQUV)	P7-1	5.41E-12	< 0.01 %
7-2 In-vessel Fuel-Coolant Interaction (AE)	P7-2	4.22E-13	< 0.01 %
8-1 Ex-vessel Fuel-Coolant Interaction (TQUV)	P8-1	2.51E-10	0.01 %
8-2 Ex-vessel Fuel-Coolant Interaction (AE)	P8-2	1.50E-11	< 0.01 %
9 Direct Containment Heating (TQUX)	P9	2.41E-11	< 0.01 %
10-1 PCV Isolation Failure (TQUV)	P10-1	2.28E-10	0.01 %
10-2 PCV Isolation Failure (AE)	P10-2	8.06E-11	< 0.01 %
11-1 Molten Core Concrete Interaction (TQUV)	P11-1	1.91E-09	0.08 %
11-2 Molten Core Concrete Interaction (AE)	P11-2	1.77E-11	< 0.01 %
12 RPV rupture (S4)	P12	1.00E-08	0.44 %
13 Containment Bypass (S3E)	P13	1.85E-08	0.82 %
14 S/P Bypass (TNQUV)	P14	1.75E-10	0.01 %
15 Direct Debris Interaction (TQUX)	P15	2.70E-09	0.12 %
16 Long Term SBO (TB in-vessel FCI)	P16	1.58E-09	0.07 %
IE at Power Level 2 PSA Sub-total		**6.49E-08**	**2.9 %**
BO-SDO-L	S1	6.69E-08	2.95 %
BO-SDO-E	S2	1.72E-09	0.08 %
DD-SDO	S3	2.00E-11	< 0.01 %
BO-SDL	S4	2.53E-10	0.01 %
BO-SDH	S5	1.03E-12	< 0.01 %
BO-SDO-E + BO-SFP_LOCA 02	S6	1.38E-09	0.06 %
IE at Shutdown Level 2 PSA Sub-total		**7.03E-08**	**3.1 %**

Table 25.15.4-6 Breakdown of Contributors to Facility Dose Band: > 1,000 mSv (2/5)

release category / sequence group / fault group	Level 3 PSA case	Frequency (/y)	% of total band frequency
BO-SFP02	F1	2.44E-08	1.08 %
BO-SFP03	F1	2.51E-10	0.01 %
BO-SFP-LOCA02	F1	2.18E-08	0.96 %
BO-SFP-LOCA03	F1	2.06E-10	0.01 %
BO-SFP02 + RC	F1+P13	1.00E-09	0.04 %
BO-SFP03 + RC	F1+P13	9.70E-12	< 0.01 %
BO-SFP-LOCA02 + RC	F1+P13	5.10E-11	< 0.01 %
BO-SFP-LOCA03 + RC	F1+P13	4.40E-14	< 0.01 %
DDS-SFP05 + RC	F2+P13	1.10E-11	< 0.01 %
DDL-SFP03	F3	7.00E-14	< 0.01 %
DDL-SFP02 + RC	F3+P13	8.80E-11	< 0.01 %
DDL-SFP03 + RC	F3+P13	4.08E-13	< 0.01 %
SFP IE Level 2 PSA sub-total		**4.79E-08**	**2.1 %**
CP*: Success sequence with containment overpressure failure (with success of VSS) with fuel perforation	CP*	1.21E-07	5.33 %
VP* : Success sequence with containment failure (with VSS failure) with fuel perforation	VP*	3.66E-08	1.61 %
B* : Success sequence with containment bypass with fuel perforation	B*	3.12E-09	0.14 %
CDTF : Cask drop onto truck bay shaft or LOOP with SGTS unsuccessful	CDTF	4.85E-09	0.21 %
Events not leading to fuel melt sub-total		**1.66E-07**	**7.3 %**
IF 2 Containment Venting (with RPV breach)	P2	4.99E-10	0.02 %
IF 4 Early Containment Failure (D/W breach)	P4	5.47E-08	2.41 %
IF 5-1 Late Containment Failure (D/W breach, core damage at early phase)	P5-1	9.79E-10	0.04 %
IF 5-2 Late Containment Failure (D/W breach, core damage at medium phase)	P5-2	1.56E-10	0.01 %
IF 5-3 Late Containment Failure (D/W breach, core damage at late phase)	P5-3	9.58E-10	0.04 %
IF 5-4 Late Containment Failure (TW-LP)	P5-4	9.01E-09	0.40 %
IF 6 Late Containment Failure (D/W breach with PCV spray success)	P6	1.61E-09	0.07 %

Table 25.15.4-6 Breakdown of Contributors to Facility Dose Band: > 1,000 mSv (3/5)

release category / sequence group / fault group	Level 3 PSA case	Frequency (/y)	% of total band frequency
IF 7-1 In-vessel FCI (TQUV)	P7-1	1.00E-13	< 0.01 %
IF 7-2 In-vessel FCI (AE)	P7-2	1.00E-13	< 0.01 %
IF 8-1 Ex-vessel FCI (TQUV)	P8-1	3.10E-11	< 0.01 %
IF 8-2 Ex-vessel FCI (AE)	P8-2	3.02E-11	< 0.01 %
IF 9 Direct Containment Heating	P9	2.16E-10	0.01 %
IF 10-1 PCV Isolation failure (core damage at early phase)	P10-1	8.06E-10	0.04 %
IF 10-2 PCV Isolation failure (core damage at late phase)	P10-2	7.07E-09	0.31 %
IF 11-1 MCCI (TQUV)	P11-1	1.87E-10	0.01 %
IF 11-2 MCCI (AE)	P11-2	2.01E-11	< 0.01 %
IF 12 RPV rupture (with LDF success)	P12	2.62E-09	0.12 %
IF 13 Containment Bypass	P13	7.54E-08	3.32 %
IF 14 S/P Bypass	P14	1.24E-07	5.47 %
IF 15 Direct Debris Interaction	P15	2.63E-08	1.16 %
IF 16 Long Term SBO	P16	1.30E-10	0.01 %
IFL 2 Containment Venting (with RPV breach)	P2	7.84E-09	0.35 %
IFL 4 Early Containment Failure (D/W breach)	P4	3.73E-08	1.64 %
IFL 5-1 Late Containment Failure (D/W breach, core damage at early phase)	P5-1	2.58E-08	1.14 %
IFL 5-2 Late Containment Failure (D/W breach, core damage at medium phase)	P5-2	2.40E-09	0.11 %
IFL 5-3 Late Containment Failure (D/W breach, core damage at late phase)	P5-3	3.28E-09	0.14 %
IFL 5-4 Late Containment Failure (TW-LP)	P5-4	2.65E-08	1.17 %
IFL 6 Late Containment Failure (D/W breach with PCV spray success)	P6	4.33E-08	1.91 %
IFL 7-1 In-vessel FCI (TQUV)	P7-1	2.38E-11	< 0.01 %
IFL 7-2 In-vessel FCI (AE)	P7-2	4.04E-12	< 0.01 %
IFL 8-1 Ex-vessel FCI (TQUV)	P8-1	3.26E-09	0.14 %
IFL 8-2 Ex-vessel FCI (AE)	P8-2	5.49E-10	0.02 %
IFL 9 Direct Containment Heating	P9	2.40E-09	0.11 %
IFL 10-1 PCV Isolation failure (core damage at early phase)	P10-1	5.13E-08	2.26 %

Table 25.15.4-6 Breakdown of Contributors to Facility Dose Band: > 1,000 mSv (4/5)

release category / sequence group / fault group	Level 3 PSA case	Frequency (/y)	% of total band frequency
IFL 10-2 PCV Isolation failure (core damage at late phase)	P10-2	5.47E-08	2.41 %
IFL 11-1 MCCI (TQUV)	P11-1	8.74E-09	0.39 %
IFL 11-2 MCCI (AE)	P11-2	5.05E-10	0.02 %
IFL 12 RPV rupture (with LDF success)	P12	4.39E-08	1.94 %
IFL 13 Containment Bypass	P13	2.74E-08	1.21 %
IFL 14 S/P Bypass	P14	2.44E-09	0.11 %
IFL 15 Direct Debris Interaction	P15	2.36E-07	10.41 %
IFL 16 Long Term SBO	P16	9.24E-12	< 0.01 %
Internal Hazards Level 2 PSAs sub-total		**8.82E-07**	**38.9 %**
ES 2 Containment Venting (TQUV no DW sprays)	P2	2.53E-09	0.11 %
ES 4 Early Containment Failure (AC)	P4	3.19E-08	1.41 %
ES 5-1 Late Containment Failure (TQUV)	P5-1	9.38E-09	0.41 %
ES 5-2 Late Containment Failure (AE)	P5-2	2.55E-08	1.12 %
ES 5-3 Late Containment Failure (AW-LP)	P5-3	4.91E-10	0.02 %
ES 5-4 late Containment Failure (TW-LP)	P5-4	7.56E-10	0.03 %
ES 6 Late Containment Failure with PCV spray (AE)	P6	3.19E-09	0.14 %
ES 7-1 In-vessel Fuel-Coolant Interaction (TQUV)	P7-1	6.49E-12	< 0.01 %
ES 7-2 In-vessel Fuel-Coolant Interaction (AE)	P7-2	5.58E-12	< 0.01 %
ES 8-1 Ex-vessel Fuel-Coolant Interaction (TQUV)	P8-1	8.12E-11	< 0.01 %
ES 8-2 Ex-vessel Fuel-Coolant Interaction (AE)	P8-2	2.19E-09	0.10 %
ES 9 Direct Containment Heating (TQUX)	P9	2.49E-10	0.01 %
ES 10-1 PCV Isolation Failure (TQUV)	P10-1	1.45E-08	0.64 %
ES 10-2 PCV Isolation Failure (AE)	P10-2	8.97E-08	3.95 %
ES 11-1 Molten Core Concrete Interaction (TQUV)	P11-1	3.05E-09	0.13 %
ES 11-2 Molten Core Concrete Interaction (AE)	P11-2	1.20E-08	0.53 %
ES 12 RPV rupture (S4)	P12	1.43E-07	6.30 %
ES 13 Containment Bypass (S3E)	P13	2.32E-07	10.23 %
ES 14 S/P Bypass (TNQUV)	P14	2.70E-08	1.19 %
ES 15 Direct Debris Interaction (TQUX)	P15	1.11E-08	0.49 %
ES 16 Long Term SBO (TB in-vessel FCI)	P16	4.28E-08	1.89 %

Table 25.15.4-6 Breakdown of Contributors to Facility Dose Band: > 1,000 mSv (5/5)

release category / sequence group / fault group	Level 3 PSA case	Frequency (/y)	% of total band frequency
SBO-SFP02	F1 + RC	2.22E-07	9.79 %
SBO-SFP03		2.82E-08	1.24 %
SBO-SFP-LOCA02		8.10E-08	3.57 %
SBO-SFP-LOCA03		1.18E-09	0.05 %
SBO-SFP-LOCA04		2.79E-09	0.12 %
SBO-SFP-LOCA05		3.22E-10	0.01 %
SDDS-SFP02	F2	3.32E-08	1.46 %
SDDS-SFP03		4.32E-09	0.19 %
SDDL-SFP02	F3 + RC	1.14E-08	0.50 %
SDDL-SFP03		1.38E-09	0.06 %
External Hazards Level 2 PSAs sub-total		**1.04E-06**	**45.7 %**
Total Frequency /y		**2.27E-06**	
BSO		1.00E-06	
Total as % of BSL		**2.3 %**	
BSL		1.00E-04	

25.15.4.2 Assessment of Individual Risk (Target 7)

The assessment against the individual risk targets is presented in Table 25.15.4-7. The summed individual risk varies between 3.24E-07 /y at 400 m and 1.66E-07 /y at 1500 m. This range is considered to bracket the distances of the nearest habitations or locations of significant public occupancy. For assessment against the risk targets, the individual risk at 1km is used as the benchmark; this is 2.06E-07 /y or 20.6 percent of the BSO and 0.206 percent of the BSL.

The individual risk breakdown at 1km is shown graphically in Figure 25.15.4-10 (excluding the hazard initiators) and Figure 25.15.4-11 (including the hazard initiators).

The individual contributors to the total individual risk at 1km are given in Table 25.15.4-8. The dominant contributors are:

- ES 13 Containment Bypass (S3E) {2.86E-08 /y},
- SBO-SFP02 Boil Off with failure of spray system and blow out panel {2.73E-08 /y},
- ES 12 RPV rupture (S4) {1.51E-08 /y},
- BO-SDO-L (Late release at RPV open or S/P bypass condition by fuel damage due to boil-off) {1.23E-08 /y},
- IF 4 Early Containment Failure (D/W breach) {1.17E-08 /y},
- ES10-2 PCV Isolation Failure (AE) {1.01E-08 /y},
- SBO-SFP_LOCA02{9.95E-09 /y},
- IF 15 S/P Bypass {9.49E-09 /y},

- IF 13 Containment Bypass{9.28E-09 /y},
- IFL 10-2 PCV Isolation failure (core damage at late phase) {6.17E-09 /y},
- IF 4 Early Containment Failure (D/W breach) {6.09E-09 /y},
- ES 16 Long Term SBO (TB in-vessel FCI) {4.64E-09 /y},
- IFL12 RPV rupture (S4) {4.63E-09 /y},
- IFL4 Early Containment Failure (AC) {4.15E-09 /y},
- SDDS-SFP02{3.93E-09 /y},
- ES 4 Early Containment Failure (AC) {3.55E-09 /y},
- SBO-SFP03 {3.46E-09 /y},
- IFL 13 Containment Bypass {3.38E-09 /y},
- BO-SFP02 {2.87E-09 /y},
- BO-SFP-LOCA02 {2.57E-09 /y},
- ES 14 S/P Bypass (TNQUV) {2.54E-09 /y}, and
- 13 Containment Bypass (S3E) {2.28E-09 /y}.

No other contributors are <1 percent of the BSO.

Table 25.15.4-7 Compiled Table for Comparison against Individual Risk Target (Target 7)

Initiating Event Group	Individual risk of fatal health effects (/y)		
	400 m	1000 m	1500 m
Internal Events at Power Level 2 PSA	9.58E-09	5.94E-09	4.71E-09
Internal Events at Shutdown Level 2 PSA	1.72E-08	1.28E-08	1.10E-08
Spent Fuel Pool Internal Events Level 2 PSA	7.57E-09	5.63E-09	4.89E-09
Events not leading to fuel melt	6.76E-09	2.05E-09	1.23E-09
Internal Hazards Level 2 PSA: Fire at Power	4.73E-08	2.99E-08	2.34E-08
Internal Hazards Level 2 PSA: Flood at Power	6.65E-08	3.39E-08	2.46E-08
External Hazards: seismic event affecting reactor at Power	1.06E-07	6.91E-08	5.50E-08
External Hazards: seismic event affecting SFP	6.32E-08	4.70E-08	4.12E-08
Total individual risk (/y):	**3.24E-07**	**2.06E-07**	**1.66E-07**
Total as % of BSO	**32.4 %**	**20.6 %**	**16.6 %**
BSO	1.00E-06		
BSL	1.00E-04		

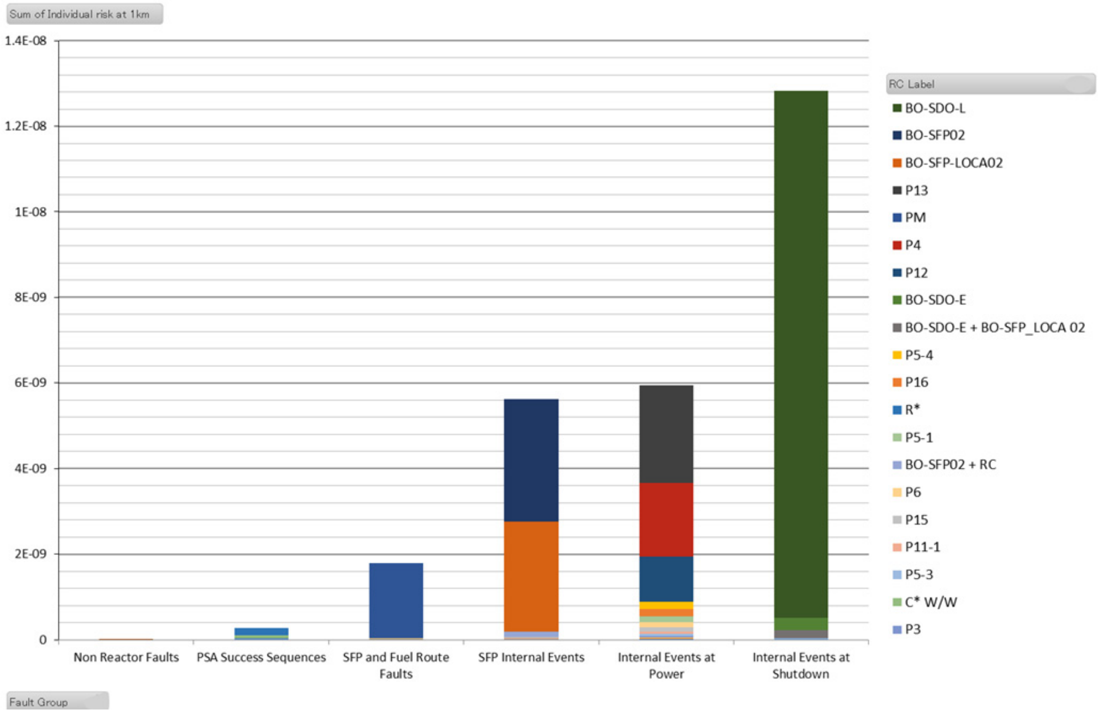


Figure 25.15.4-10 Contributors to Individual Risk at 1 km (for Target 7) Excluding Hazard Initiators

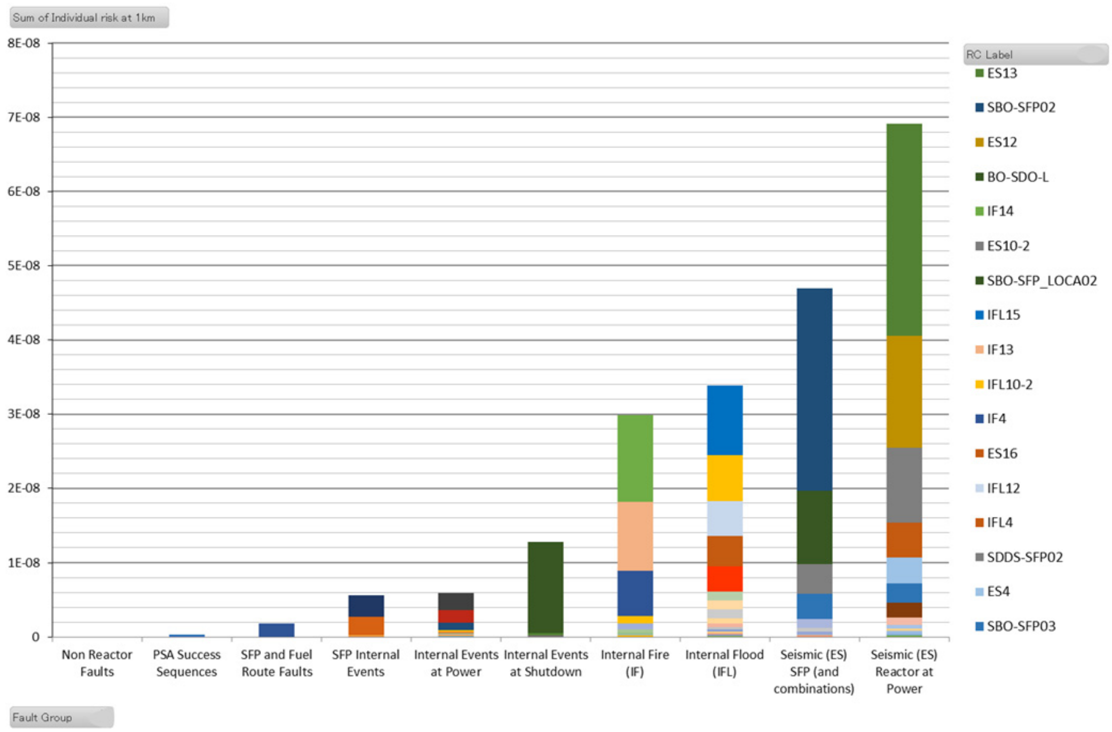


Figure 25.15.4-11 Contributors to Individual Risk at 1 km (for Target 7) Including Hazard Initiators (Internal and External)

Table 25.15.4-8 Breakdown of Contributors to Individual Risk at 1 km (1/7)

release category / sequence group / fault group	Level 3 PSA case	Frequency (/y)	Individual risk at 1 km (/y)	% of total risk
1 Containment Leakage from D/W - failed RPV (TQUV)	P1	3.84E-08	8.79E-14	< 0.01 %
2 Containment Venting (TQUV no DW sprays)	P2	1.05E-09	4.28E-12	< 0.01 %
3 Filtered Containment Venting (TQUV)	P3	1.31E-07	3.43E-11	0.02 %
4 Early Containment Failure (AC)	P4	1.54E-08	1.71E-09	0.83 %
5-1 Late Containment Failure (TQUV)	P5-1	4.39E-09	1.33E-10	0.06 %
5-2 Late Containment Failure (AE)	P5-2	6.33E-11	4.94E-12	< 0.01 %
5-3 Late Containment Failure (AW-LP)	P5-3	3.47E-10	5.03E-11	0.02 %
5-4 late Containment Failure (TW-LP)	P5-4	3.93E-09	1.73E-10	0.08 %
6 Late Containment Failure with PCV spray (AE)	P6	4.21E-09	1.18E-10	0.06 %
7-1 In-vessel Fuel-Coolant Interaction (TQUV)	P7-1	5.41E-12	5.04E-13	< 0.01 %
7-2 In-vessel Fuel-Coolant Interaction (AE)	P7-2	4.22E-13	5.12E-14	< 0.01 %
8-1 Ex-vessel Fuel-Coolant Interaction (TQUV)	P8-1	2.51E-10	3.95E-12	< 0.01 %
8-2 Ex-vessel Fuel-Coolant Interaction (AE)	P8-2	1.50E-11	1.26E-12	< 0.01 %
9 Direct Containment Heating (TQUX)	P9	2.41E-11	1.31E-12	< 0.01 %
10-1 PCV Isolation Failure (TQUV)	P10-1	2.28E-10	5.26E-12	< 0.01 %
10-2 PCV Isolation Failure (AE)	P10-2	8.06E-11	9.09E-12	< 0.01 %
11-1 Molten Core Concrete Interaction (TQUV)	P11-1	1.91E-09	5.71E-11	0.03 %
11-2 Molten Core Concrete Interaction (AE)	P11-2	1.77E-11	1.59E-12	< 0.01 %
12 RPV rupture (S4)	P12	1.00E-08	1.05E-09	0.51 %
13 Containment Bypass (S3E)	P13	1.85E-08	2.28E-09	1.10 %
14 S/P Bypass (TNQUV)	P14	1.75E-10	1.65E-11	0.01 %
15 Direct Debris Interaction (TQUX)	P15	2.70E-09	1.08E-10	0.05 %
16 Long Term SBO (TB in-vessel FCI)	P16	1.58E-09	1.71E-10	0.08 %
IE at Power Level 2 PSA Sub-total			**5.94E-09**	**2.9 %**

Table 25.15.4-8 Breakdown of Contributors to Individual Risk at 1 km (2/7)

release category / sequence group / fault group	Level 3 PSA case	Frequency (/y)	Individual risk at 1 km (/y)	% of total risk
BO-SDO-L	S1	6.69E-08	1.23E-08	5.96 %
BO-SDO-E	S2	1.72E-09	2.97E-10	0.14 %
DD-SDO	S3	2.00E-11	3.13E-12	< 0.01 %
BO-SDL	S4	2.53E-10	2.40E-11	0.01 %
BO-SDH	S5	1.03E-12	3.56E-14	< 0.01 %
BO-SDO-E + BO-SFP_LOCA 02	S6	1.38E-09	1.87E-10	0.09 %
IE at Shutdown Level 2 PSA Sub-total			**1.28E-08**	**6.2 %**
BO-SFP02	F1	2.44E-08	2.87E-09	1.39 %
BO-SFP03	F1	2.51E-10	2.95E-11	0.01 %
BO-SFP-LOCA02	F1	2.18E-08	2.57E-09	1.24 %
BO-SFP-LOCA03	F1	2.06E-10	2.42E-11	0.01 %
BO-SFP02 + RC	F1+P13	1.00E-09	1.23E-10	0.06 %
BO-SFP03 + RC	F1+P13	9.70E-12	1.19E-12	< 0.01 %
BO-SFP-LOCA02 + RC	F1+P13	5.10E-11	6.26E-12	< 0.01 %
BO-SFP-LOCA03 + RC	F1+P13	4.40E-14	5.40E-15	< 0.01 %
DDS-SFP05 + RC	F2+P13	1.10E-11	1.36E-12	< 0.01 %
DDL-SFP03	F3	7.00E-14	4.78E-15	< 0.01 %
DDL-SFP02 + RC	F3+P13	8.80E-11	9.25E-12	< 0.01 %
DDL-SFP03 + RC	F3+P13	4.08E-13	4.29E-14	< 0.01 %
SFP IE Level 2 PSA Sub-total			**5.63E-09**	**2.7 %**
C : Success sequence by containment venting (containment intact)	C	6.63E-04	8.40E-12	< 0.01 %
CP : Success sequence with containment overpressure failure (with success of VSS)	CP	1.36E-03	1.72E-11	0.01 %
VP : Success sequence with containment failure (with VSS failure)	VP	8.19E-07	1.04E-14	< 0.01 %
B : Success sequence with containment bypass	B	1.51E-05	2.68E-13	< 0.01 %
R* : Success sequence by RHR operation (containment intact) with fuel perforation	R*	1.78E-04	1.69E-10	0.08 %
C* : Success sequence by containment venting (containment intact) with fuel perforation	C*	5.39E-06	4.53E-11	0.02 %

Table 25.15.4-8 Breakdown of Contributors to Individual Risk at 1 km (3/7)

release category / sequence group / fault group	Level 3 PSA case	Frequency (/y)	Individual risk at 1 km (/y)	% of total risk
CP* : Success sequence with containment overpressure failure (with success of VSS) with fuel perforation	CP*	1.21E-07	2.34E-11	0.01 %
VP* : Success sequence with containment failure (with VSS failure) with fuel perforation	VP*	3.66E-08	4.29E-12	< 0.01 %
B* : Success sequence with containment bypass with fuel perforation	B*	3.12E-09	1.74E-13	< 0.01 %
PM : Success sequence by water makeup in SFP	PM	1.32E-01	1.74E-09	0.84 %
FS : Fuel Handling Accident with SGTS successful	FS	3.19E-04	4.75E-12	< 0.01 %
FH : Fuel Handling Accident with SGTS unsuccessful	FH	6.51E-05	9.70E-13	< 0.01 %
HLPS : Heavy Load (e.g. cask) drop into spent fuel pool with SGTS successful	HLPS	1.43E-08	2.21E-14	< 0.01 %
HLPF : Heavy Load (e.g. cask) drop into spent fuel pool with SGTS unsuccessful	HLPF	3.93E-10	6.07E-16	< 0.01 %
CDPS : Cask drop onto cask pit or preparation pit with SGTS successful	CDPS	2.07E-05	3.20E-11	0.02 %
CDPF : Cask drop onto cask pit or preparation pit with SGTS unsuccessful	CDPF	7.60E-07	1.17E-12	< 0.01 %
CDTS : Cask drop onto truck bay shaft or LOOP with SGTS successful	CDTS	2.67E-08	2.90E-12	< 0.01 %
CDTF : Cask drop onto truck bay shaft or LOOP with SGTS unsuccessful	CDTF	4.85E-09	5.27E-13	< 0.01 %
NO1 : Offgas System Failure unmitigated case	NO1+H/U Rupture	1.00E-05	3.90E-15	< 0.01 %
Events not leading to fuel melt Sub-total			**2.05E-09**	**1.0 %**
IF 1 Containment Leakage from D/W (with RPV breach)	P1	2.33E-08	5.33E-14	< 0.01 %
IF 2 Containment Venting (with RPV breach)	P2	4.99E-10	2.04E-12	< 0.01 %
IF 3 Filtered Containment Venting (with RPV breach)	P3	1.67E-07	4.39E-11	0.02 %
IF 4 Early Containment Failure (D/W breach)	P4	5.47E-08	6.09E-09	2.95 %

Table 25.15.4-8 Breakdown of Contributors to Individual Risk at 1 km (4/7)

release category / sequence group / fault group	Level 3 PSA case	Frequency (/y)	Individual risk at 1 km (/y)	% of total risk
IF 5-1 Late Containment Failure (D/W breach, core damage at early phase)	P5-1	9.79E-10	2.97E-11	0.01 %
IF 5-2 Late Containment Failure (D/W breach, core damage at medium phase)	P5-2	1.56E-10	1.22E-11	0.01 %
IF 5-3 Late Containment Failure (D/W breach, core damage at late phase)	P5-3	9.58E-10	1.39E-10	0.07 %
IF 5-4 Late Containment Failure (TW-LP)	P5-4	9.01E-09	3.97E-10	0.19 %
IF 6 Late Containment Failure (D/W breach with PCV spray success)	P6	1.61E-09	4.50E-11	0.02 %
IF 7-1 In-vessel FCI (TQUV)	P7-1	1.00E-13	9.31E-15	< 0.01 %
IF 7-2 In-vessel FCI (AE)	P7-2	1.00E-13	1.21E-14	< 0.01 %
IF 8-1 Ex-vessel FCI (TQUV)	P8-1	3.10E-11	4.89E-13	< 0.01 %
IF 8-2 Ex-vessel FCI (AE)	P8-2	3.02E-11	2.54E-12	< 0.01 %
IF 9 Direct Containment Heating	P9	2.16E-10	1.18E-11	0.01 %
IF 10-1 PCV Isolation failure (core damage at early phase)	P10-1	8.06E-10	1.86E-11	0.01 %
IF 10-2 PCV Isolation failure (core damage at late phase)	P10-2	7.07E-09	7.98E-10	0.39 %
IF 11-1 MCCI (TQUV)	P11-1	1.87E-10	5.58E-12	< 0.01 %
IF 11-2 MCCI (AE)	P11-2	2.01E-11	1.81E-12	< 0.01 %
IF 12 RPV rupture (with LDF success)	P12	2.62E-09	2.77E-10	0.13 %
IF 13 Containment Bypass	P13	7.54E-08	9.28E-09	4.49 %
IF 14 S/P Bypass	P14	1.24E-07	1.17E-08	5.67 %
IF 15 Direct Debris Interaction	P15	2.63E-08	1.06E-09	0.51 %
IF 16 Long Term SBO	P16	1.30E-10	1.41E-11	0.01 %
Internal Hazards: Fire Level 2 PSA Sub-total			**2.99E-08**	**14.5 %**
IFL 1 Containment Leakage from D/W (with RPV breach)	P1	7.71E-09	1.76E-14	< 0.01 %
IFL 2 Containment Venting (with RPV breach)	P2	7.84E-09	3.20E-11	0.02 %
IFL 3 Filtered Containment Venting (with RPV breach)	P3	1.16E-06	3.04E-10	0.15 %
IFL 4 Early Containment Failure (D/W breach)	P4	3.73E-08	4.15E-09	2.01 %

Table 25.15.4-8 Breakdown of Contributors to Individual Risk at 1 km (5/7)

release category / sequence group / fault group	Level 3 PSA case	Frequency (/y)	Individual risk at 1 km (/y)	% of total risk
IFL 5-1 Late Containment Failure (D/W breach, core damage at early phase)	P5-1	2.58E-08	7.84E-10	0.38 %
IFL 5-2 Late Containment Failure (D/W breach, core damage at medium phase)	P5-2	2.40E-09	1.87E-10	0.09 %
IFL 5-3 Late Containment Failure (D/W breach, core damage at late phase)	P5-3	3.28E-09	4.76E-10	0.23 %
IFL 5-4 Late Containment Failure (TW-LP)	P5-4	2.65E-08	1.17E-09	0.57 %
IFL 6 Late Containment Failure (D/W breach with PCV spray success)	P6	4.33E-08	1.21E-09	0.59 %
IFL 7-1 In-vessel FCI (TQUV)	P7-1	2.38E-11	2.22E-12	< 0.01 %
IFL 7-2 In-vessel FCI (AE)	P7-2	4.04E-12	4.91E-13	< 0.01 %
IFL 8-1 Ex-vessel FCI (TQUV)	P8-1	3.26E-09	5.14E-11	0.02 %
IFL 8-2 Ex-vessel FCI (AE)	P8-2	5.49E-10	4.61E-11	0.02 %
IFL 9 Direct Containment Heating	P9	2.40E-09	1.31E-10	0.06 %
IFL 10-1 PCV Isolation failure (core damage at early phase)	P10-1	5.13E-08	1.18E-09	0.57 %
IFL 10-2 PCV Isolation failure (core damage at late phase)	P10-2	5.47E-08	6.17E-09	2.99 %
IFL 11-1 MCCI (TQUV)	P11-1	8.74E-09	2.61E-10	0.13 %
IFL 11-2 MCCI (AE)	P11-2	5.05E-10	4.54E-11	0.02 %
IFL 12 RPV rupture (with LDF success)	P12	4.39E-08	4.63E-09	2.24 %
IFL 13 Containment Bypass	P13	2.74E-08	3.38E-09	1.64 %
IFL 14 S/P Bypass	P14	2.44E-09	2.30E-10	0.11 %
IFL 15 Direct Debris Interaction	P15	2.36E-07	9.46E-09	4.58 %
IFL 16 Long Term SBO	P16	9.24E-12	1.00E-12	< 0.01 %
Internal Hazards: Flood Level 2 PSA Sub-total			**3.39E-08**	**16.4 %**
ES 1 Containment Leakage from D/W - failed RPV (TQUV)	P1	2.12E-08	4.85E-14	< 0.01 %
ES 2 Containment Venting (TQUV no DW sprays)	P2	2.53E-09	1.03E-11	< 0.01 %
ES 3 Filtered Containment Venting (TQUV no DW sprays)	P3	6.14E-08	1.61E-11	0.01 %
ES 4 Early Containment Failure (AC)	P4	3.19E-08	3.55E-09	1.72 %

Table 25.15.4-8 Breakdown of Contributors to Individual Risk at 1 km (6/7)

release category / sequence group / fault group	Level 3 PSA case	Frequency (/y)	Individual risk at 1 km (/y)	% of total risk
ES 5-1 Late Containment Failure (TQUV)	P5-1	9.38E-09	2.85E-10	0.14 %
ES 5-2 Late Containment Failure (AE)	P5-2	2.55E-08	1.99E-09	0.96 %
ES 5-3 Late Containment Failure (AW-LP)	P5-3	4.91E-10	7.11E-11	0.03 %
ES 5-4 late Containment Failure (TW-LP)	P5-4	7.56E-10	3.33E-11	0.02 %
ES 6 Late Containment Failure with PCV spray (AE)	P6	3.19E-09	8.91E-11	0.04 %
ES 7-1 In-vessel Fuel-Coolant Interaction (TQUV)	P7-1	6.49E-12	6.04E-13	< 0.01 %
ES 7-2 In-vessel Fuel-Coolant Interaction (AE)	P7-2	5.58E-12	6.77E-13	< 0.01 %
ES 8-1 Ex-vessel Fuel-Coolant Interaction (TQUV)	P8-1	8.12E-11	1.28E-12	< 0.01 %
ES 8-2 Ex-vessel Fuel-Coolant Interaction (AE)	P8-2	2.19E-09	1.84E-10	0.09 %
ES 9 Direct Containment Heating (TQUX)	P9	2.49E-10	1.36E-11	0.01 %
ES 10-1 PCV Isolation Failure (TQUV)	P10-1	1.45E-08	3.34E-10	0.16%
ES 10-2 PCV Isolation Failure (AE)	P10-2	8.97E-08	1.01E-08	4.89 %
ES 11-1 Molten Core Concrete Interaction (TQUV)	P11-1	3.05E-09	9.12E-11	0.04 %
ES 11-2 Molten Core Concrete Interaction (AE)	P11-2	1.20E-08	1.08E-09	0.52 %
ES 12 RPV rupture (S4)	P12	1.43E-07	1.51E-08	7.31 %
ES 13 Containment Bypass (S3E)	P13	2.32E-07	2.86E-08	13.85 %
ES 14 S/P Bypass (TNQUV)	P14	2.70E-08	2.54E-09	1.23 %
ES 15 Direct Debris Interaction (TQUX)	P15	1.11E-08	4.45E-10	0.22 %
ES 16 Long Term SBO (TB in-vessel FCI)	P16	4.28E-08	4.64E-09	2.25 %
External Hazards: seismic event (reactor at Power) Level 2 PSA Sub-total			**6.91E-08**	**33.5 %**
SBO-SFP02	F1+RC	2.22E-07	2.73E-08	13.22 %
SBO-SFP03		2.82E-08	3.46E-09	1.68 %
SBO-SFP-LOCA02		8.10E-08	9.95E-09	4.82 %
SBO-SFP-LOCA03		1.18E-09	1.45E-10	0.07 %
SBO-SFP-LOCA04		2.79E-09	3.43E-10	0.17 %
SBO-SFP-LOCA05		3.22E-10	3.95E-11	0.02 %

Table 25.15.4-8 Breakdown of Contributors to Individual Risk at 1 km (7/7)

release category / sequence group / fault group	Level 3 PSA case	Frequency (/y)	Individual risk at 1 km (/y)	% of total risk
SDDS-SFP02	F2	3.32E-08	3.93E-09	1.90%
SDDS-SFP03		4.32E-09	5.11E-10	0.25%
SDDL-SFP02	F3+RC	1.14E-08	1.20E-09	0.58%
SDDL-SFP03		1.38E-09	1.45E-10	0.07%
External Hazards: seismic event (SFP) Level 2 PSA Sub-total			**4.70E-08**	**22.8 %**
Total individual risk at 1km /y			**2.06E-07**	
Total as % of BSO			**20.6 %**	
		BSO	1.00E-06	
		BSL	1.00E-04	

25.15.4.3 Assessment of Societal Risk (Target 9)

The assessment against the societal risk targets is presented in Table 25.15.4-9. The summed frequency allocated to Target 9 is 2.10E-06 /y. This is a factor of 21 above the BSO or 21 percent of the BSL. This frequency is dominated by the hazard initiators which, together, contribute 1.62E-06 /y. The remainder contribute 4.88E-07 /y (a factor of 4.88 above the BSO) or 4.88 percent of the BSL.

The frequency breakdown is shown graphically in Figure 25.15.4-12 (excluding the hazard initiators) and Figure 25.15.4-13 (including the hazard initiators).

The individual contributors to the frequency assessed as above the societal risk criterion of Target 9 are shown in Table 25.15.4-10 are very similar to those given for the highest facility dose band in Table 25.15.4-6. This is because the frequency dominant release categories in the highest dose band are considered to contribute to the frequency of exceeding the Target 9 threshold. Therefore, the dominant groups are similar to those given for the Facility Dose Bands discussed above with the exception of the three PSA ‘success’ sequence groups: CP*, VP* and B* which are not included.

In the Level 3 PSA methodology, developed for events leading to fuel melt, the mean values (The mean values typically represent the 70 to 75th percentile results.) of fatalities are used for comparison against the Target 9 threshold. We call this the direct comparison approach. If the sum of the mean values is >100 then all the frequency is allocated as above the threshold. Similarly, if the sum of the mean value is < 100, then none of the frequency is allocated as above the threshold unless a qualitative consideration of potential short term onsite fatalities is considered to lift the conditional consequences above the threshold.

Table 25.15.4-9 Compiled Table for Comparison against Societal Risk Target (Target 9)

Initiating Event Group	Frequency above threshold (/y)	% of BSO
Internal Events at Power Level 2 PSA	6.49E-08	64.9 %
Internal Events at Shutdown Level 2 PSA	7.03E-08	70.3 %
Spent Fuel Pool Internal Events Level 2 PSA	4.79E-08	47.9 %
Events not leading to fuel melt	0.00E+00	0.0 %
Internal Hazards Level 2 PSA: Fire at Power	3.05E-07	305 %
Internal Hazards Level 2 PSA: Flood at Power	5.78E-07	578 %
External Hazards: seismic event affecting reactor at Power	6.51E-07	651 %
External Hazards: seismic event affecting SFP	3.86E-07	386 %
Total frequency above threshold (/y):	**2.10E-06**	**2100 %**
	Total as % of BSL	**21.0 %**
BSO	1.00E-07	
BSL	1.00E-05	

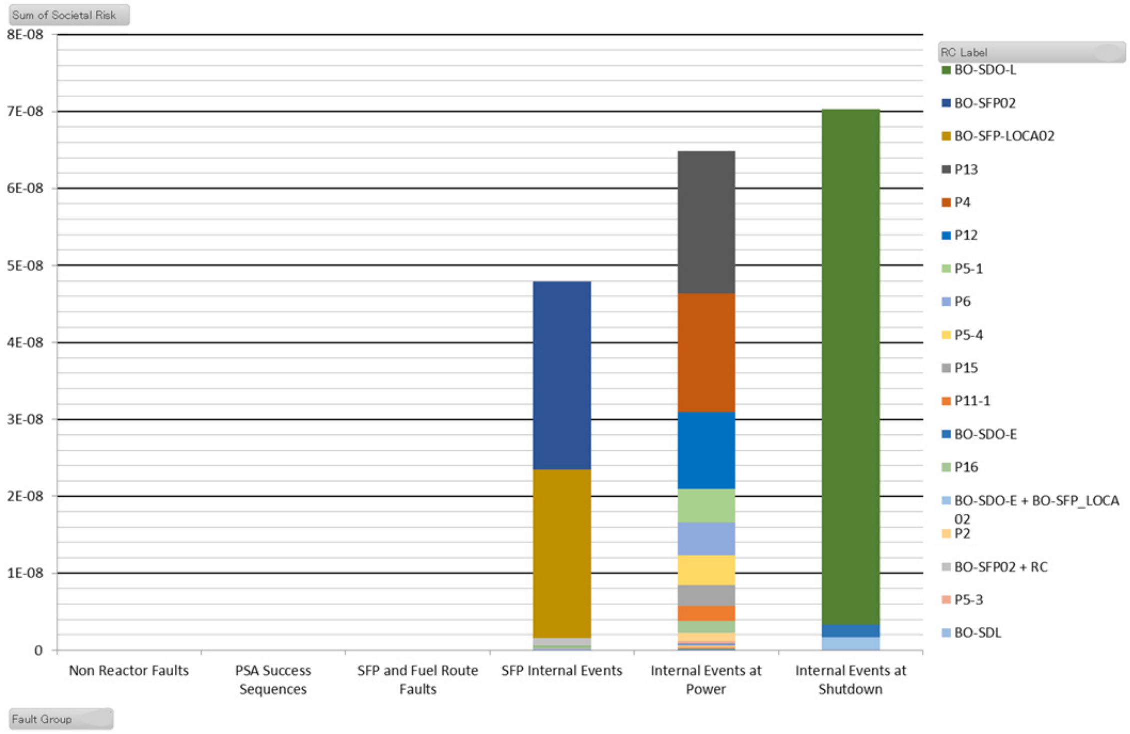


Figure 25.15.4-12 Contributors to Frequency of Exceeding the Target 9 Criterion Excluding Hazard Initiators

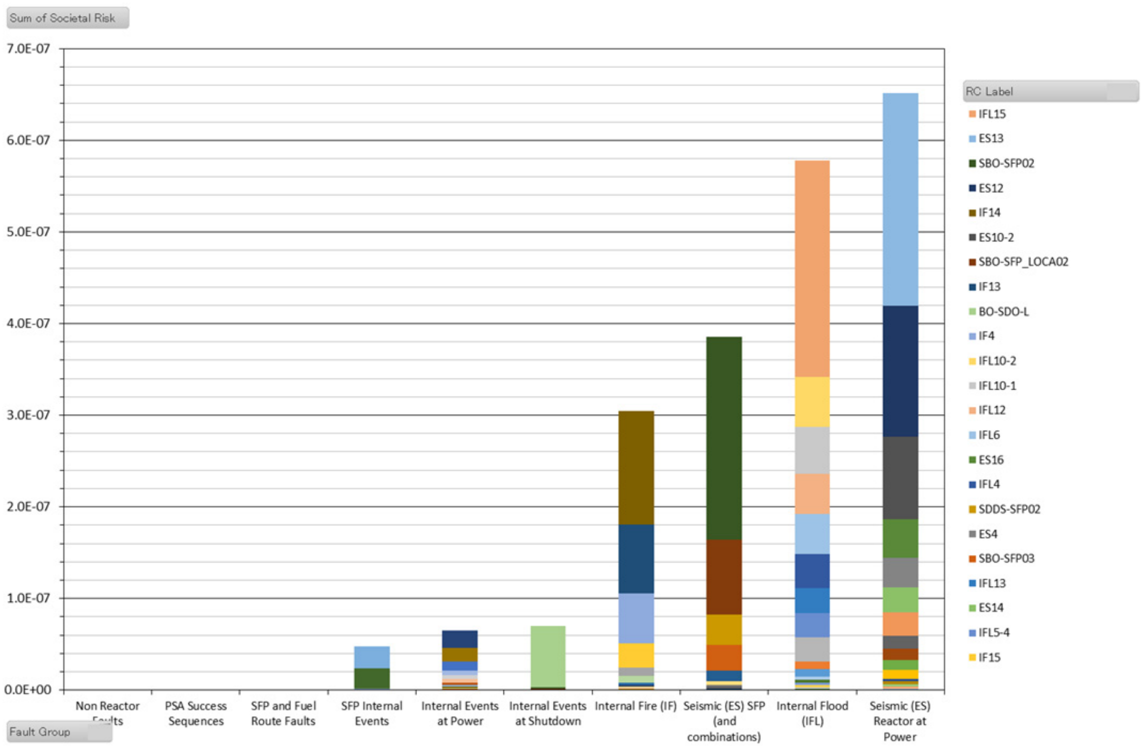


Figure 25.15.4-13 Contributors to Frequency of Exceeding the Target 9 Criterion Including Hazard Initiators (Internal and External)

Table 25.15.4-10 Breakdown of Contributors to Frequency above Societal Risk Threshold (1/5)

release category / sequence group / fault group	Level 3 PSA case	Frequency (/y)	% of total frequency
2 Containment Venting (TQUV no DW sprays)	P2	1.05E-09	0.05 %
4 Early Containment Failure (AC)	P4	1.54E-08	0.73 %
5-1 Late Containment Failure (TQUV)	P5-1	4.39E-09	0.21 %
5-2 Late Containment Failure (AE)	P5-2	6.33E-11	< 0.01 %
5-3 Late Containment Failure (AW-LP)	P5-3	3.47E-10	0.02 %
5-4 Late Containment Failure (TW-LP)	P5-4	3.93E-09	0.19 %
6 Late Containment Failure with PCV spray (AE)	P6	4.21E-09	0.20 %
7-1 In-vessel Fuel-Coolant Interaction (TQUV)	P7-1	5.41E-12	< 0.01 %
7-2 In-vessel Fuel-Coolant Interaction (AE)	P7-2	4.22E-13	< 0.01 %
8-1 Ex-vessel Fuel-Coolant Interaction (TQUV)	P8-1	2.51E-10	0.01 %
8-2 Ex-vessel Fuel-Coolant Interaction (AE)	P8-2	1.50E-11	< 0.01 %
9 Direct Containment Heating (TQUX)	P9	2.41E-11	< 0.01 %
10-1 PCV Isolation Failure (TQUV)	P10-1	2.28E-10	0.01 %
10-2 PCV Isolation Failure (AE)	P10-2	8.06E-11	< 0.01 %
11-1 Molten Core Concrete Interaction (TQUV)	P11-1	1.91E-09	0.09 %
11-2 Molten Core Concrete Interaction (AE)	P11-2	1.77E-11	< 0.01 %
12 RPV rupture (S4)	P12	1.00E-08	0.48 %
13 Containment Bypass (S3E)	P13	1.85E-08	0.88 %
14 S/P Bypass (TNQUV)	P14	1.75E-10	0.01 %
15 Direct Debris Interaction (TQUX)	P15	2.70E-09	0.13 %
16 Long Term SBO (TB in-vessel FCI)	P16	1.58E-09	0.08 %
IE at Power Level 2 PSA Sub-total		**6.49E-08**	**3.1 %**
BO-SDO-L	S1	6.69E-08	3.19 %
BO-SDO-E	S2	1.72E-09	0.08 %
DD-SDO	S3	2.00E-11	< 0.01 %
BO-SDL	S4	2.53E-10	0.01 %
BO-SDH	S5	1.03E-12	< 0.01 %
BO-SDO-E + BO-SFP_LOCA 02	S6	1.38E-09	0.07 %
IE at Shutdown Level 2 PSA Sub-total		**7.03E-08**	**3.3 %**

Table 25.15.4-10 Breakdown of Contributors to Frequency above Societal Risk Threshold (2/5)

release category / sequence group / fault group	Level 3 PSA case	Frequency (/y)	% of total frequency
BO-SFP02	F1	2.44E-08	1.16 %
BO-SFP03	F1	2.51E-10	0.01 %
BO-SFP-LOCA02	F1	2.18E-08	1.04 %
BO-SFP-LOCA03	F1	2.06E-10	0.01 %
BO-SFP02 + RC	F1+P13	1.00E-09	0.05 %
BO-SFP03 + RC	F1+P13	9.70E-12	< 0.01 %
BO-SFP-LOCA02 + RC	F1+P13	5.10E-11	< 0.01 %
BO-SFP-LOCA03 + RC	F1+P13	4.40E-14	< 0.01 %
DDS-SFP05 + RC	F2+P13	1.10E-11	< 0.01 %
DDL-SFP03	F3	7.00E-14	< 0.01 %
DDL-SFP02 + RC	F3+P13	8.80E-11	< 0.01 %
DDL-SFP03 + RC	F3+P13	4.08E-13	< 0.01 %
SFP IE Level 2 PSA Sub-total		**4.79E-08**	**2.3 %**
none			
Events not leading to fuel melt Sub-total		**0.00E+00**	**0.00 %**
IF 2 Containment Venting (with RPV breach)	P2	4.99E-10	0.02 %
IF 4 Early Containment Failure (D/W breach)	P4	5.47E-08	2.61 %
IF 5-1 Late Containment Failure (D/W breach, core damage at early phase)	P5-1	9.79E-10	0.05 %
IF 5-2 Late Containment Failure (D/W breach, core damage at medium phase)	P5-2	1.56E-10	0.01 %
IF 5-3 Late Containment Failure (D/W breach, core damage at late phase)	P5-3	9.58E-10	0.05 %
IF 5-4 Late Containment Failure (TW-LP)	P5-4	9.01E-09	0.43 %
IF 6 Late Containment Failure (D/W breach with PCV spray success)	P6	1.61E-09	0.08 %
IF 7-1 In-vessel FCI (TQUV)	P7-1	1.00E-13	< 0.01 %
IF 7-2 In-vessel FCI (AE)	P7-2	1.00E-13	< 0.01 %
IF 8-1 Ex-vessel FCI (TQUV)	P8-1	3.10E-11	< 0.01 %
IF 8-2 Ex-vessel FCI (AE)	P8-2	3.02E-11	< 0.01 %
IF 9 Direct Containment Heating	P9	2.16E-10	0.01 %

Table 25.15.4-10 Breakdown of Contributors to Frequency above Societal Risk Threshold (3/5)

release category / sequence group / fault group	Level 3 PSA case	Frequency (/y)	% of total frequency
IF 10-1 PCV Isolation failure (core damage at early phase)	P10-1	8.06E-10	0.04 %
IF 10-2 PCV Isolation failure (core damage at late phase)	P10-2	7.07E-09	0.34 %
IF 11-1 MCCI (TQUV)	P11-1	1.87E-10	0.01 %
IF 11-2 MCCI(AE)	P11-2	2.01E-11	< 0.01 %
IF 12 RPV rupture (with LDF success)	P12	2.62E-09	0.12 %
IF 13 Containment Bypass	P13	7.54E-08	3.59 %
IF 14 S/P Bypass	P14	1.24E-07	5.90 %
IF 15 Direct Debris Interaction	P15	2.63E-08	1.25 %
IF 16 Long Term SBO	P16	1.30E-10	0.01 %
IFL 2 Containment Venting (with RPV breach)	P2	7.84E-09	0.37 %
IFL 4 Early Containment Failure (D/W breach)	P4	3.73E-08	1.78 %
IFL 5-1 Late Containment Failure (D/W breach, core damage at early phase)	P5-1	2.58E-08	1.23 %
IFL 5-2 Late Containment Failure (D/W breach, core damage at medium phase)	P5-2	2.40E-09	0.11 %
IFL 5-3 Late Containment Failure (D/W breach, core damage at late phase)	P5-3	3.28E-09	0.16 %
IFL 5-4 Late Containment Failure (TW-LP)	P5-4	2.65E-08	1.26 %
IFL 6 Late Containment Failure (D/W breach with PCV spray success)	P6	4.33E-08	2.06 %
IFL 7-1 In-vessel FCI (TQUV)	P7-1	2.38E-11	< 0.01 %
IFL 7-2 In-vessel FCI (AE)	P7-2	4.04E-12	< 0.01 %
IFL 8-1 Ex-vessel FCI (TQUV)	P8-1	3.26E-09	0.16 %
IFL 8-2 Ex-vessel FCI (AE)	P8-2	5.49E-10	0.03 %
IFL 9 Direct Containment Heating	P9	2.40E-09	0.11 %
IFL 10-1 PCV Isolation failure (core damage at early phase)	P10-1	5.13E-08	2.44 %
IFL 10-2 PCV Isolation failure (core damage at late phase)	P10-2	5.47E-08	2.61 %
IFL 11-1 MCCI (TQUV)	P11-1	8.74E-09	0.42 %
IFL 11-2 MCCI(AE)	P11-2	5.05E-10	0.02 %
IFL12 RPV rupture (with LDF success)	P12	4.39E-08	2.09 %
IFL 13 Containment Bypass	P13	2.74E-08	1.31 %

Table 25.15.4-10 Breakdown of Contributors to Frequency above Societal Risk Threshold (4/5)

release category / sequence group / fault group	Level 3 PSA case	Frequency (/y)	% of total frequency
IFL 14 S/P Bypass	P14	2.44E-09	0.12 %
IFL 15 Direct Debris Interaction	P15	2.36E-07	11.24 %
IFL 16 Long Term SBO	P16	9.24E-12	< 0.01 %
Internal Hazards Level 2 PSAs Sub-total		**8.82E-07**	**42.0 %**
ES 4 Early Containment Failure (AC)	P4	3.19E-08	1.52%
ES 5-1 Late Containment Failure (TQUV)	P5-1	9.38E-09	0.45%
ES 5-2 Late Containment Failure (AE)	P5-2	2.55E-08	1.21%
ES 5-3 Late Containment Failure (AW-LP)	P5-3	4.91E-10	0.02%
ES 5-4 Late Containment Failure (TW-LP)	P5-4	7.56E-10	0.04%
ES 6 Late Containment Failure with PCV spray (AE)	P6	3.19E-09	0.15%
ES 7-1 In-vessel Fuel-Coolant Interaction (TQUV)	P7-1	6.49E-12	< 0.01 %
ES 7-2 In-vessel Fuel-Coolant Interaction (AE)	P7-2	5.58E-12	< 0.01 %
ES 8-1 Ex-vessel Fuel-Coolant Interaction (TQUV)	P8-1	8.12E-11	< 0.01 %
ES 8-2 Ex-vessel Fuel-Coolant Interaction (AE)	P8-2	2.19E-09	0.10%
ES 9 Direct Containment Heating (TQUX)	P9	2.49E-10	0.01 %
ES 10-1 PCV Isolation Failure (TQUV)	P10-1	1.45E-08	0.69%
ES 10-2 PCV Isolation Failure (AE)	P10-2	8.97E-08	4.27%
ES 11-1 Molten Core Concrete Interaction (TQUV)	P11-1	3.05E-09	0.15%
ES 11-2 Molten Core Concrete Interaction (AE)	P11-2	1.20E-08	0.57%
ES 12 RPV rupture (S4)	P12	1.43E-07	6.81%
ES 13 Containment Bypass (S3E)	P13	2.32E-07	11.05%
ES 14 S/P Bypass (TNQUV)	P14	2.70E-08	1.29%
ES 15 Direct Debris Interaction (TQUX)	P15	1.11E-08	0.53%
ES 16 Long Term SBO (TB in-vessel FCI)	P16	4.28E-08	2.04%
SBO-SFP02	F1 + RC	2.22E-07	10.57%
SBO-SFP03		2.82E-08	1.34%
SBO-SFP-LOCA02		8.10E-08	3.86%
SBO-SFP-LOCA03		1.18E-09	0.06%
SBO-SFP-LOCA04		2.79E-09	0.13%
SBO-SFP-LOCA05		3.22E-10	0.02%

Table 25.15.4-10 Breakdown of Contributors to Frequency above Societal Risk Threshold (5/5)

release category / sequence group / fault group	Level 3 PSA case	Frequency (/y)	% of total frequency
SDDS-SFP02	F2	3.32E-08	1.58%
SDDS-SFP03		4.32E-09	0.21%
SDDL-SFP02	F3 + RC	1.14E-08	0.54%
SDDL-SFP03		1.38E-09	0.07%
External Hazards Level 2 PSAs Sub-total		**1.04E-06**	**49.4 %**
Total Frequency /y		**2.10E-06**	
Total as % of BSO		**2100 %**	
BSO		1.00E-07	
Total as % of BSL		**21.0 %**	
BSL		1.00E-05	

25.15.5 PSA Key Insights

The UK ABWR PSA has been developed in order to support the design development and to demonstrate that the risks associated with the design and operation of the UK ABWR is ALARP, such that it can be concluded that there are no further reasonable practicable improvements to be made to the generic plant design during GDA.

The identification of risk characteristics and insights has been performed separately for each fault group since the results have different levels of detail and levels of uncertainty. The differing level of uncertainty is largely due to the availability of details in design information during the design phase. Assessed fault groups include:

- Internal Initiating Events at Power,
- Internal Initiating Events during Shutdown,
- Internal Initiating Events for the Spent Fuel Pool,
- Internal Fire for at Power (refined result), Shutdown and the Spent Fuel Pool,
- Internal Flooding for at Power (refined result), Shutdown and the Spent Fuel Pool,
- Seismic Hazard for at Power, Shutdown and the Spent Fuel Pool,
- Non Reactor Faults including Fuel Route, and
- Other Hazards (Turbine Missile, Tornado Missile and Accidental Aircraft Impact and External Flooding/Water-based Biological Fouling).

The general selection process can be described as a systematic review of each fault group to be performed in support of the ALARP demonstration. Risk significant characteristics are summarised from the individual PSA results. This includes a systematic review identifying any plant vulnerabilities and major modelling uncertainties in the individual PSA or other improvements that could be made in the plant design or operation to reduce the plant risk.

Identification of Risk Contributors and Options

The first step is to identify risk significant scenarios and to identify potential options for plant safety improvement. The PSA results are examined from the perspective of ALARP. The following steps were performed for each PSA group:

(1) The individual PSA results were systematically reviewed to identify any potential plant vulnerabilities:

  (a) The results for various fault groups were reviewed to determine if there are any dominant risk contributors in the Minimum Cutsets (MCS) or sequence results.

  (b) The risk importance measures were reviewed to identify basic events contributing greater than 10 percent (Fussell-Vesely) or has a risk achievement worth (RAW) greater than or equal to a factor of 2 increase to the overall plant risk (CDF, LRF or Level 3 PSA results).

  (c) The results involve a low order cutset where possible inadequate defence-in-depth or diversity is indicated. Additional sensitivity analysis were performed, if necessary, to provide further supporting information.

(2) A systematic review of the completed sensitivity analysis were performed for each fault group that involved potential plant or design improvements. Additional sensitivity analysis were performed, if necessary, to provide further supporting information.

(3) Fault assumptions and uncertainties were systematically reviewed to determine and avoid any potential misleading insights being derived. The goal was to find potential plant or design improvements which could help reduce the overall plant risk and uncertainty.

(4) The initiating event frequencies of internal events faults were assessed to identify their significance and the ability of the UK ABWR to respond to them.

From the above process, PSA key insights were identified from the individual PSA insights for further ALARP consideration, such as identification of options and assessment on risk reduction [Ref-25.11]. Table 25.15.5-1 to Table 25.15.5-7 show the PSA key insights from each PSA. Furthermore, Table 25.15.5-8 shows the PSA key insights based on integrated results of importance analyses, which are discussed in Section 25.15.2 and Section 25.15.3.

Note: External flooding and water-based biological fouling are assessed as a sensitivity analysis because hazard assessment of external flooding and water-based biological fouling may not be available in the GDA time frame. Although the risk insights are provided in Section 25.11.3.3, ALARP for the hazards will be examined at the post GDA phase.

Note: The key insights from other hazards PSA (Turbine missile, Tornado missile and Accidental air craft impact) are provided in Table 25.15.5-8 of [Ref-25.1].

Note: The evaluation of the Non-Reactor Faults shows that both BSL and BSO limits are met for all dose levels as described in Section 25.15.4. Therefore, non-reactor faults are not discussed in this section.

These insights were used for assessment on ALARP described in Section 25.16.

Table 25.15.5-1 Key Insights from IEAP PSA

Insights and key items in insights from IEAP PSA
Transients contributing 2 percent or more to the total IEAP CDF are: - Loss of feedwater, - Loss of condenser heat sink, and - General transients.
Three principle contributors were found to contribute to manual shutdown risk: - Loss of two or more divisions of RCW/RSW; - Loss of HNCW and - Loss of TB HVAC. Loss of two or more divisions of RCW/RSW has been identified as a significant contributor to CDF. The loss of two divisions indicates that there is only a single division of ECCS available to provide core and containment cooling.
LOCA accidents contribute a total of 14 percent of the IEAP CDF. The leading LOCA contributor to CDF is Excessive LOCA/Vessel Rupture which is assumed to result in both core damage and containment failure.
RCIC operability under loss of Class 1 AC power. RCIC operation up to 8 hours is credited in the PSA
Heat Capacity Temperature Limit (HCTL) This limit leads to a shorter time margin for operator action
Testing of the HPCF and LPCF injection valves may lead to an ISLOCA.
The PRV hatches are assumed to fail given RPV failure at high pressure. Containment performance analysis indicates the hatches do not fail if the lower drywell is pre-flooded.
COPS setpoint A rupture disk opening at twice the pressure of PCV design pressure (2Pd) is mounted on downstream of PCV isolation valves from W/W. If the COPS setpoint is changed to be lower than 2Pd, S/P water temperature will be lower. In this case, HPCF pump suctioning from S/P water may continuously operate.
CRD injection late The MUWC/CRD system can inject water to RPV therefore it has the potential to injection after HCTL.
Loss of Offsite Power grid and weather related is one the most significant of the so-called internal events initiators are initiated from outside the plant.
Switchyard centred LOOP is an important contributor to total IEAP CDF
ISLOCA associated with the RHR suction line is a dominant contributor to IEAP CDF and has the potential to bypass the containment. The current results are governed by the assumption that spurious operation of the isolation valves in this line can expose the RHR piping to full reactor pressure, causing a large break.
HPCF accident operability under high temperature/pressure conditions in the containment. Can reduce the potential for containment failure to cause loss of core cooling.
Long term operability of SRVs for depressurisation
Make-up water condensate system(MUWC) and injection / makeup to CST
ATWS scenarios Investigate prohibiting RPV depressurisation vs Low pressure injection
RPV water level control in BOC/ISLOCA The failure of RPV water level control leads to the submergence of the R/B.

Table 25.15.5-2 Key Insights from Shutdown PSA

Insights and key items in insights from Internal Events Shutdown PSA
CCF to open all 16 SRVs (passive safety valve function) contributes approximately 28 percent to internal events Shutdown FDF and has a RAW equivalent to over a factor of 2 increase to the overall plant risk.
"Fail to open" events of the feedwater-A line isolation check valves B21-F051A(B) and RHR-A return line check valve E11-F006A impact RPV injection from RHR-A, FLSS and FLSR.
"Fail to open" events of the FLSS RPV injection line check valves E71-F024 and E71-F025, and "Plug" event of FLSS RPV injection line manual valve E71-F026 impact RPV injection from FLSS and FLSR.
"Plug" event of FLSS/FLSR SFP spray sparger E71-FD009 impacts SFP injection from FLSS and FLSR.
Loss of Class 1 AC during POS C involving loss of RHR-B or its support systems, and loss of HPCF-B, LPRF-B, MUWC-B (all dependent on division 2 AC power)
Loss of the operating RHR frontline system during POS C (IE) contributes approximately 17 percent to the total internal events shutdown FDF.
LOCA (mechanical) above normal water level during POS B-2. The frequency of this IE group is dominated by that of heavy load drop. The overhead cranes for UK ABWR are equipped with interlocks. The heavy load drop frequency is dominated by HEP
___-HFE-SC-FL (Failure of manual initiation of FLSS) contributes 18 percent to the total internal events shutdown FDF.
___-HFE-SC-MU (Failure of manual initiation of MUWC) contributes 13 percent to the total internal events shutdown FDF.
FLSR-SD_ST (FLSR (Mobile Injection Facility) Unavailability (Line-up before IE)) contributes 17 percent to the total internal events shutdown FDF.
Fire protection system for SFP injection reduces the internal events shutdown FDF by approximately 20 percent.
"Recovery of components is not credited except for offsite power and the systems failed by human error(s) as an internal events shutdown IE."
Credit to recovery of failed components reduces the internal events shutdown FDF by approximately 50 percent.
"Some of the PCV boundaries, (e.g., personnel air lock), are open throughout the outage period even when the PCV head is closed (except for the PCV leak test during POS D)."
The PCV function of confining fission products is not credited in the shutdown PSA, resulting in the LRF equal to FDF in all the accident sequences involving fuel damage in the reactor.

Table 25.15.5-3 Key Insights from SFP PSA

Insights and key items in insights from Internal Events SFP PSA
Credit on FLSS spray in Level 1 PSA reduces the SFP FDF approximately 89 percent.
In the SFP PSA, fuel damage is defined as “Fuel Uncovery”. However, this criterion includes conservatism because spent fuel can be cooled and the fuel damage can be mitigated by spray after the fuel uncovery. In the sensitivity case, the spray function is credited in the Level 2 PSA to avoid the fuel heating up and large release.
The largest risk contribution PDS of SFP Level 2 PSA is Boil-off (larger than 99 percent of the total SFP FDF). Investigate assumption of no-recovery action.
Containment failure sequences caused by reactor accident have some potential to lead SFP faults (so-called reactor challenge). SFP PSA uses conditional probability for each containment failure inducing effect such as steaming, strain/deformation and hydrogen effect. Investigate conditional probabilities used.
“Plug” event of FPC manual valve G41 MV_-PG-___-F017B and F019B impact SFP injection from FPC.
“Plug” event of FLSS/FLSR SFP spray sparger E71-FD009 impacts SFP injection from FLSS and FLSR.
“Loss of function” events of electric components and its support system such as M/C, P/C, MCC, circuit breaker, transformer and damper of HVAC impact SFP cooling and injection.
Loss of Class 1 AC bus during POS A is the highest contributor: approximately 49 percent of the total internal events FDF for SFP.
Operating FPC-A fails due to various causes due to the IE including Loss of Class 1 AC bus.
LOCA at CUW outside PCV during POS D is the second highest contributor: approximately 19 percent of the total internal events FDF for SFP.
No safety system is credited whose components are located in the R/B. This effectively fails: FPC-A (B), MUWC-A (B, C), RHR SFP mode and Fire Protection system.
FLSR is important in all POS for SFP PSA.
It is assumed FLSR is required to be connected to the injection point before an IE during POS C so that the redundancy of mitigation systems is improved under the condition Class 1 divisions 1 and 3 systems are in maintenance. This is consistent with shutdown PSA modelling.
Due to the partially degraded redundancy of the Class 1 systems (In POS D, one train of FLSS is assumed to be in maintenance). The post-initiator HFE. Manual initiation of FLSS contributes
Dependency assumption for Multiple Human Failure Events.
For Level 1 PSA, zero dependency is considered between Human Failure Events due to time available from an initiating event to fuel damage is over 30 hours. Additional cognition and actions are expected before fuel uncovery which is the definition of fuel damage because the time margin exceeds the length of a shift rotation.

Table 25.15.5-4 Key Insights from Internal Fire PSA

(Refined at Power Analysis and Scoping Analyses of Shutown and SFP)

Insights and key items in insights from Internal Fire at Power PSA
Fire risk originating in RB contributes nearly 70 percent to the refined Internal Fire at Power LRF. The primary reason is that fires originating in the RB Divisions I and II electrical rooms contribute the most of any plant area. This is in large part due to existence of a large number of ignition sources (cabinets) and critical cables. Furthermore, locations of some ignition sources and critical cables were unknown and thus the worst locations were assumed.
Fire risk originating in HxB contributes approximately 1 percent to the refined Internal Fire at Power CDF and LRF.
Fire risk originating in TB contributes approximately 7 percent and 4 percent to the refined Internal Fire at Power CDF and LRF, respectively.
Fire risk originating in CB contributes approximately 18 percent and 7 percent to the refined Internal Fire at Power CDF and LRF, respectively.
Random “fail to open” events of the feedwater-A line isolation check valves B21-F051A, B21-52A and RHR-A return line check valve E11-F006A impact RPV injection from RHR-A, FLSS and FLSR.
Random events of non-electrical components, i.e., orifice E71-D004, check valves E71-F023/F024/F025, manual valves E71-F022/F026, in the common RPV injection line for FLSS have RAWs equivalent to over a factor of 2 increase to the overall plant risk.
Fire risk originating in BB contributes approximately 11 percent and 16 percent to the refined Internal Fire at Power CDF and LRF, respectively.
Fire-induced spurious operations resulting from hot shorts in BB contribute marginally to the refined Fire PSA risk profile.
Fire-induced spurious containment venting (MSO 4SU) contributes approximately 5 percent to the refined at Power LRF.
Fire-induced spurious containment spray by FLSS (MSO ABWR3) contributes approximately 10 percent to the CDF and LRF for internal fire at power. This MSO fails wetwell venting as well as FLSS itself.
Loss of EDG or BBG due to fire-induced spurious overloading contributes approximately 34 percent to the refined at Power LRF. Internal Fire PSA considers loss of an EDG or BBG due to overloading caused by fire-induced spurious operations. The failure modes includes spurious closure of a circuit breaker between a Class 1 M/C (or a Class 2 P/C) and a pump with bypassing sequencing logic or load shedding logic. Such scenarios are characterised by Multiple Spurious Operations (MSO).
The top 3 cutsets of the Internal Fire at Power CDF involve the whole room damage scenario for the RB-BB connecting service tunnel.
Loss or spurious failure of RVI transmitters is conservatively assumed to occur with a conditional probability of 1.0 given the associated RVI sensing line is exposed to fire impact.
Spurious open of the S/P drain line secondary stop valve, combined with spurious open of one of the RHR S/P drain line primary stop valves, results in MSO 2M which is diversion of S/P water to the S/P water drainage system. This MSO contributes approximately 10 percent to the refined Internal Fire at Power LRF. Specifically, random spurious open of the S/P drain line secondary stop valve has a RAW equivalent to over a factor of 2 increase to the overall plant risk.
Fire risk originating in the BB contributes significantly to the FDF of the reactor during Shutdown POS B-2, POS C and SFP POS F. This is in large part due to fire-induced spurious overfill of the reactor well or SFP from FLSS which causes internal flooding in the RB (if unisolated).
Fire risk originating in the TB contributes more than 20 percent to the Internal Fire Shutdown FDF and Internal Fire SFP FDF, based on the scoping analysis.
Fire risk originating in the CB contributes approximately 20 percent to the Internal Fire SFP FDF.

Table 25.15.5-5 Key Insights from Internal Flood PSA

(Refined at Power Analysis and Scoping Analyses of Shutown and SFP) (1/2)

Insights and key items in insights from Internal Flood at Power PSA
Internal flood risk originating in the Heat Exchanger Building (Hx/B) contributes approximately 17 percent and 3 percent to the refined Internal Flood at Power CDF and LRF, respectively.
Internal flood risk originating in the Control Building (C/B) contributes approximately 4 percent and 6 percent to the refined Internal Flood at Power CDF and LRF, respectively.
Internal flood risk originating in the Turbine Building (T/B) contributes approximately 3 percent and 7 percent to the refined Internal Flood at Power CDF and LRF, respectively.
Internal flood risk originating in the Back-up Building (B/B) contributes less than 1 percent to the refined Internal Flood at Power CDF and LRF, respectively.
Random “fail to open” events of the RHR pump (B) outlet check valve and minimum flow line check valve and random “inadvertent close” events of suction valve from S/P, RHR Hx (B) outlet valve, and minimum flow valve impact RPV injection from RHR-B. Additionally, random “loss of function” events of class 1 DC bus B also impacts RPV injection from RHR-B.
Random “fail to open” events of the RHR pump (C) outlet check valve and minimum flow line check valve, random “inadvertent close” events of suction valve from S/P, RHR Hx (C) outlet valve, and minimum flow valve, and random “plugging” events of RHR Hx (C), RHR Hx (F), and RHR pump (C) local cooling unit impact RPV injection from RHR-C.
Unsealed ceiling penetrations in the Reactor Building (R/B) B3F elevation pump rooms allow propagation from higher elevation flood areas to enter the pump rooms and potentially affect multiple safety divisions.
Internal flood risk originating in the R/B contributes approximately 80 percent to the refined Internal Flood at Power CDF and LRF.
The assumed component critical height is lower than the height of the curbs surrounding floor penetrations and the curbs beneath most doors that separate rooms. As a result, the vast majority of flood areas that accumulate sufficient flood depth to propagate to another flood area already have all components in that flood area failed.
The spatial geometry between flood sources, HELB barriers, and components within flood areas is unknown.
For the piping systems in the UK ABWR design, only design parameters were available for use in determining high-energy status and leak rates.
The capping of floor drains, due to the lack of divisional separation in the drainage system, provides an inability to mitigate and control flood propagation, which otherwise could reduce the impact of spray scenarios in the source area and flood scenarios in propagated areas.
Automatic switchover from the Condensate Storage Tank (CST) to Suppression Pool (S/P) as a suction source for the RCIC and HPCF pumps does not have a manual inhibit function
Only water-tight doors are located in the ECCS pump rooms at the B3F elevation of the R/B.
The importance of events contributing to RPS failure is more important for the Level 2 PSA as ATWS sequences have no Level 2 mitigation (i.e., FLSS is failed due to a consequential LOCA).
The importance of events contributing to failure of the Vapour Suppression System (VSS) as a result of S/P drainage is more important for the Level 2 PSA. Containment isolation is failed due to the break resulting in S/P drainage; therefore, there are no Level 2 mitigation systems available. These scenarios lead to the S/P bypass release category.
Random “fail to open” events of the feedwater-A line isolation check valves and RHR-A return line check valve impact RPV injection from RHR-A, FLSS and FLSR. These events have RAWs equivalent to over a factor of 2 increase to the overall plant risk.
Random events of non-electrical components, i.e., orifice, check valves, manual valves, in the common RPV injection line for FLSS have RAWs equivalent to over a factor of 2 increase to the overall plant risk.
Random BB DC buse failures have RAWs equivalent to over a factor of 2 increase to the overall plant risk.

Table 25.15.5-5 Key Insights from Internal Flood PSA

(Refined at Power Analysis and Scoping Analyses of Shutown and SFP) (2/2)

Insights and key items in insights from Internal Flood at Power PSA
LOCA events inside the PCV contributes nearly 90 percent to the internal flood FDF of Shutdown POS B-2.
LOCA at the Feedwater-A line inside the PCV contributes approximately 50 percent to the internal flood FDF of Shutdown POS C.

Table 25.15.5-6 Key Insights from Seismic PSA

Insights and key items in insights from Seismic PSA
The largest contribution to CDF and LRF for reactor at Power Seismic PSA Level 1 and Level 2 PSA comes from 0.8 to 0.9 g PGA (Peak Ground Acceleration).
In the Level 1 PSA, at low ground acceleration below 0.6g, SBO sequences are dominant; most of which are caused by EDG support system failure such as RSW pumps and DG Light Oil Tanks, whose seismic capacities are small. Above 0.6 g, the contribution of LOCA initiating scenarios gets larger with increasing seismic hazard level.
In the SFP Level 1 PSA, at low ground acceleration level below 0.3 g, damage sequences are dominated by the scenarios in LOOP condition. In this scenario, the restart of FPC is credited since there is still sufficient water in SFP. At the seismic hazard above 0.3 g, FPC cannot be used due to SFP water sloshing leading to low water level in skimmer surge tank. In the 0.3 to 0.5 g seismic hazard interval, although Class 1 AC power is available in many cases, no make-up function can be credited by seismic category 3 system such as MUWC. Make up only by FLSS or FLSR can be credited in the seismic hazard interval. At seismic hazards above 0.5 g, the sequences in SBO condition dominate the total scenarios followed by SFP small leak sequence.
In the SFP Level 2 PSA, Boil-off scenarios represent most of the LRF sequences through all bands of seismic hazard. The boil off scenarios can be classified into two groups; one caused by the failure of FPC and another caused by reactor BOC. With increasing seismic hazard level, the contribution of scenarios with reactor BOC becomes large.
SFP: Loss of FPC cooling.
Reactor: TB group (TB, TBU and TBPN).
Seismically induced failure of ceramic insulator has the highest FV importance.
Seismically induced failure of Class 1 instrumentation rack or Class 1 control panel failure impacts the CDF/LRF because it causes loss of Safety Class 1 systems.
Seismically induced failure of Main Steam Isolation Valve (MSIV) impacts the CDF/LRF because it causes BOC with no mitigation measures.
Seismically induced failure of FLSS/FLSR piping and spray plug impact the FDF/LRF for SFP because it causes no injection from FLSS and FLSR.
Seismically induced failure of RCCV (Reinforced Concrete Containment Vessel) impacts the FDF/LRF for SFP because it causes loss of SFP inventory without mitigation measures.
Seismically induced failure of Class 1 cable tray impacts the FDF/LRF for SFP because it causes loss of Class 1 AC power.
Seismically induced failure of Control Building (C/B) impacts the FDF/LRF for SFP because it causes loss of control of Class 1 safety measures.
The total CDF of the seismic shutdown PSA for the reactor is 4.2E-08 /y. This is about 6 percent of that of the at Power condition. The percentage of shutdown risk contribution is reasonable because of duration of the shutdown outage, the absence of reactivity control accidents, the absence of seismic specific initiating events in shutdown, and the relatively low decay heat level compared to the at Power condition.
Impact of criteria for high PIF (Performance Influence Factor) was studied by a sensitivity analysis, and shown to have a large impact on FDF and LRF for SFP
In the Level 2 PSA, the contribution of TB sequence is large where the seismic hazard level is below 0.4 g. In high hazard levels above 0.4 g, S3E and AE sequences represent a large portion of LRF. Most of S3E scenarios are originated in the BOC sequence at the CUW line in the Level 1 PSA and this scenario has no mitigation in Level 2 PSA event tree. AE sequences are originated from the group of LOCA inside containment scenarios.

Table 25.15.5-7 Key Insights from Fuel Route PSA

Insights and key items in insights from Fuel Route PSA
Release frequency of Cask Drop onto truck bay shaft (CD) scenario is 1.0E-09 /y order of magnitude.
Release frequency of LOOP during cask handling (LOOP-DR) scenario is 1.0E-08 /y order of magnitude.
Release frequency of fuel drop (FD) scenario is 3.84E-04 /y.
Release frequency of Cask Drop onto spent fuel storage pool (CD-SFP) is 1.48E-08 /y and that for Cask Drop onto cask pit or preparation pit (CD-CP) 2.15E-05 /y.
Cask handling times (i.e.25 times per year) is conservatively assumed.

Table 25.15.5-8 Key Insights from Integrated Results of Importance Analyses (1/2)

Insights and key items in insights from the integrated results of importance analyses
• (Ceramic Insulator with high F-V Importance) For seismic PSA, the contribution of TB sequences is large for total CDF/LRF because the most fragile component for offsite grid and distribution system is "ceramic insulator" and offsite power is lost due to the failure of ceramic insulator as representative component. The fragility data is obtained from EPRI report [Ref-25.111].
• (FLSS/FLSR with high F-V Importance) At the seismic hazard above 0.3 g, FPC is not credited for SFP cooling due to SFP water sloshing leading to low water level in skimmer surge tank. In the 0.3 to 0.5 g seismic hazard interval, although Class 1 AC power is available in many cases, no make-up function is credited from a seismic category 3 system such as MUWC. Make up only by FLSS or FLSR can be credited in the seismic hazard interval. In this case, FLSS/FLSR have a high F-V importance.
(Loss of a Class 1 AC bus (IE) with high F-V Importance) Loss of Class 1 AC bus due to human error significantly contributes Shutdown and SFP risk. The design options associated with this event that were considered important to achieving risk reduction across multiple fault groups were evaluated in [Ref-25.11].
• (CCF for SRVs with high F-V Importance) Fires originating in the Back-up Building (BB) contribute the greatest fire risk of any plant area. This is in large part due to the potential plant impact due to spurious operation, such as Safety Relief Valve (SRV) opening or containment vent valve opening/isolating.
(SRVs failure to open with high F-V Importance) Failure of SRVs to open is modelled with two events, including one for Shutdown/SFP (secondary impact) and one for full power, with the Shutdown event being the dominant contributor. The conservatisms in the model related to this event for Shutdown are provided below. • One of the conservatisms is use of generic failure probabilities which do not distinguish active relief valve function and passive safety valve function. A sensitivity study using a more appropriate generic failure probability of safety valve function confirmed that this event is no longer significant]. Furthermore, the CCF probability for 16 out of 16 SRVs failure is conservatively calculated using the CCF parameters for 8 SRVs. The actual CCF probability failing all 16 SRVs would be much lower, which indicates the risk contribution is over-calculated in the present PSA.
(Small LOCA in the drain and instrumentation tap or RVI piping with high F-V Importance) For the seismic event, RPV piping failure contributes to the large release risk significantly. If RVI pipings simultaneously fails due to the seismic event, automatic initiation of mitigation systems is not available. The countermeasure against the simultaneous loss of RVI piping is discussed in [Ref-25.11].
• (CUW piping with high F-V Importance) In high seismic hazard levels above 0.4 g, S3E and AE sequences represent a large portion of reactor at Power seismic LRF. Most of S3E scenarios are originated in the BOC sequence at the CUW line in the Level 1 PSA and this scenario has no mitigation in the Level 2 PSA event tree.

Table 25.15.5-8 Key Insights from Integrated Results of Importance Analyses (2/2)

Insights and key items in insights from the integrated results of importance analyses
• (Failure of valves on the RPV injection line by FLSS/FLSR with high RAW Importance) These events are most important for Shutdown, but also show up as significant for IEAP, Fire and Flood PSAs.
(Failure of valves on the SFP spray sparger by FLSS/FLSR with high RAW Importance) This event impacts SFP injection from FLSS and FLSR, and failure of this sparger shows up as risk significant in the Shutdown PSA and SFP PSA.
(Loss of DC power buses in B/B with high RAW Importance) Failure of these buses impacts RPV/PCV injection from FLSS. Failure of these buses was significant based on the unrefined internal events at power Fire PSA; however, this insight is not applicable to the refined Fire PSA results. On the other hand, failure of these buses shows up as significant in the Flood PSA.
(Failure of Class 1 control panel, instrumentation rack and FLSS/FLSR piping with high RAW Importance) The failure of the Class 1 control panel, instrumentation rack, FLSS/FLSR piping contributes to the Seismic risk.

25.16 Summary of Use of PSA in ALARP Assessment

The ABWR has evolved through the utilisation of PSA in international common engineering and specific projects. The UK ABWR has been developed based on the international design evolution of the ABWR extended to encompass the concept of ALARP.

25.16.1 Hitachi-GE's GDA Design Approach including ALARP Consideration

In the Hitachi-GE's GDA design approach, PSA was performed in conjunction with the UK ABWR GDA design activities. That is, a risk-informed design approach has been applied to the Hitachi-GE GDA design development. The UK ABWR PSA integral design role has assisted in the implementation of various design enhancements, including:

- diversity of the Class 1 C&I systems,
- diversity of the room cooling systems,
- diversity of the injection systems,
- prevention of seismically-induced simultaneous failure of reactor vessel sensing lines,
- re-routing of critical cables, and
- raising the heights of critical transmitters.

The UK ABWR PSA was also used in the development of both passive and automatic plant systems that provide protection against internal and external hazards. Also, the PSA findings led to mitigating actions for primary functions required within 30 minutes being automated for design basis transients and accidents, as well as manual back up for all functions being provided to minimise the risk of an event progressing to core or fuel damage.

In addition to the design enhancements in the development of PSA, the overall results of the individual PSAs have been assessed to identify any aspects of the plant where further design detail or plant operational activities might result in further risk reduction, and to confirm that the overall plant design is ALARP.

25.16.2 Additional Measures for ALARP Consideration based on PSA Insights

The ALARP assessment consists of a comprehensive review involving multiple factors. The review was performed by first characterizing the scenarios that contribute to risk and then identifying the plant features and UK ABWR PSA modelling important to the ALARP evaluation. As described in Section 25.15.5, to characterise the scenarios that contribute to risk, the UK ABWR PSA results were systematically reviewed to identify any potential plant vulnerabilities that resulted from dominant risk contributors in the MCS or sequence results, basic events with large risk importance measures, or inadequate defence-in-depth or diversity. In addition to the review of the UK ABWR PSA results, a systematic review of the completed sensitivity studies, hazard assumptions, and uncertainties was performed to find potential plant or design improvements that could help reduce the overall plant risk and uncertainty. Lastly, Hitachi-GE Engineering team was consulted to determine if there were any additional options that have not already been identified in the UK ABWR PSA results in the GDA that should also be considered in the ALARP assessment.

To identify the plant features and PSA modelling important to the ALARP assessment, the results from the characterization of scenarios that contribute to risk were reviewed and the plant characteristics that support risk reduction were identified. Judgement was applied to each scenario examined to identify any potential improvements in reliability and/or defence-in-depth for plant vulnerabilities or risk significant contributions. A review of the baseline and alternate options was performed using the criteria discussed in Section 25.15.5, and the review considered not only the limitations of the PSA but also the cumulative impact of the alternative options calculated across all fault groups. The ALARP review for each fault group was provided against the risk insights as shown in Table 25.15.5-1 to Table 25.15.5-8. The detail discussions and results are described in “Topic Report on Use of PSA in ALARP Assessment” [Ref-25.11].

Identification of risk characteristics, plant features, and PSA modelling important to the ALARP assessment was performed separately for each PSA fault group since the results have different levels of detail and levels of uncertainty. Following the ALARP assessment for each fault group, a cross-discipline review was performed on the combined PSA results to identify any design options and/or ALARP action items that were present in multiple fault studies. The purpose of this review was to recognise those design options or ALARP action items that may have more of a significant impact than others. This ALARP review [Ref-25.11] was provided against the risk insights as shown in Table 25.15.5-8.

Per the integrated review against the risk insights as shown in Table 25.15.5-8 [Ref-25.11], the following ALARP actions appear to be the most significant:

- The ALARP action item regarding the development of procedures and training to minimise human errors associated with FLSS and FLSR was relevant to both the Internal Events Shutdown PSA and the Internal Events SFP PSA. Failure of manual initiation of FLSS contributes approximately 18 percent to the total Internal Events Shutdown FDF, and without makeup from FLSS, a LOCA at Clean Up Water outside the Primary Containment Vessel during Plant Operating State D contributes approximately 19 percent to the total Internal Events SFP PSA. Additionally, FLSR unavailability contributes 17 percent to the total Internal Events Shutdown FDF and less than 10 percent to the Internal Events SFP PSA. Development of appropriate procedures and training related to human errors associated with FLSS and FLSR would reduce the overall plant risk for multiple faults groups.
- Failure to open the FLSS RPV injection line such as check valves E71-F024, F025 has high RAW. This failure is discussed in Table 4.3.5-1 of [Ref-25.11] for Shutdown.
- Loss of Class 1 AC bus due to human error significantly contributes Shutdown and SFP risk. The design options associated with this event that were considered important to achieving risk reduction across multiple fault groups were evaluated in [Ref-25.11].
- Another important event affecting uncertainty is failure of a SRVs to open. This is modelled with two events, including one for Shutdown/SFP and one for full power, with the Shutdown event being the dominant contributor. This failure is discussed in Table 25.15.5-9, which indicates the risk contribution is over-calculated in the present PSA.

Furthermore, additional significant actions based on the overall impact on individual important hazards include the following:

- ISLOCA associated with the RHR suction line is a dominant contributor to CDF and is included in the BYPASS release category. The current results are governed by the assumption that spurious operation of the isolation valves in this line can expose the RHR piping to full reactor pressure, causing a large break. Increasing the burst pressure of the large bore piping for RHR and HPCF suction should be evaluated to see if eliminating large breaks of this piping would demonstrate ALARP.
- Loss of Class 1 AC bus during POS A is the highest contributor: approximately 49 percent of the total internal events FDF for SFP. To reduce the risk, the design improvement is planned to install the interlock which prevents operators/workers from open this circuit breaker whilst AC bus is energised. This includes development of appropriate recovery action procedure and its training.

- Fire risk originating in BB contributes approximately 46 percent and 50 percent to the Internal Fire at Power CDF and LRF, respectively. Fire-induced spurious operations resulting from hot shorts in BB contribute significantly to the Fire PSA risk profile. Detailed design will consider countermeasure for spurious operation, such as introducing shorting switches in the circuit design of hardwired backup systems in BB.
- In high seismic hazard levels including dominant hazard interval, S3E and AE sequences with loss of Class 1 AC represent a large portion of LRF for reactor. In the detailed design phase, appropriate improvement of aseismic design of related SSCs such as EDGs, EDG's support systems (RSW or LOT) and CUW piping will be considered.
- The FPC failure due to sloshing has large risk contribution: approximately 34 percent of the total seismic FDF of SFP. To reduce the risk, aseismic design will be improved to consider sloshing in the detail design phase.
- Internal flood risk originating in the R/B contributes approximately 79 percent and 89 percent to the Internal Flood at Power CDF and LRF, respectively. Several recommendations to demonstrate ALARP [Ref-25.12] for internal flood scenarios originating in the R/B are shown below:
  - Protections of check valves for RPV injection by FLSS that are impacted by high energy line break scenarios in the main steam tunnel,
  - Celing penetrations in rooms on R/B B3F elevation,
  - Optimisation of component critical height,
  - Appropriate drain routing such as uncapped floor drains,
  - Manual inhibit function to switchover from the CST to S/P,
  - Water-tight doors between the different safety divisions at the upper elevations of the R/B B3F, and
  - Containment isolation for S/P bypass scenarios.

  The resolution of the items presented will likely have a beneficial impact on risk reduction for the R/B internal flood scenarios.

A complete list of the incorporated design features and identified options considered for the ALARP assessment is given in the various subsections of Section 4 of [Ref-25.11]. Subsections of 4.2 to 4.9 of [Ref-25.11] have two tables that list:

(1) Insights for each Fault Group Meeting ALARP Considerations.

(2) ALARP Recommendations based on each Fault Group.

According to the systematic ALARP evaluation presented in these tables, there are no further reasonably practical measures which can be taken in GDA to reduce risks further. Recommendation for reasonably practical measures which should be considered for further study beyond GDA are presented in the ALARP Recommendations based on each fault group table. Such remaining recomendations to the post GDA phase are controlled by Hitachi-GE IT system. Therefore it is concluded that the UK ABWR design in GDA complies with the principle of ALARP, and that for GDA no further reasonably practical measures to reduce risks need be considered.

25.16.3 ALARP Consideration of Mitigation Systems for Beyond Design Basis and Severe Accident Analysis

PCSR Chapter 28 (ALARP Evaluation) presents the high level approach taken for demonstrating ALARP across all aspects of the design and operation. It presents an overview of how the UK ABWR design has evolved, the further options that have been considered across all technical areas resulting in a number of design changes and how these contribute to the overall ALARP case.

ALARP considerations for Beyond Design Basis and Severe Accident Analysis are provided in PCSR Chapter 26 and summarised in this section.

The beyond design basis analysis shows this group of fault sequences do not lead to melting or considerable damage of the core so that no significant environmental release of any radioactive material occurs. It demonstrates that there are no “cliff edge” effects near the cut-off frequency of design basis faults and that the risks are ALARP.

Many design enhancements have been incorporated in the ABWR design and consistent with the ALARP principle, the UK ABWR design has been further reinforced to improve the resilience against severe accidents following the Fukushima Dai-ichi accident. In the GDA process, there were principally eight topics considered for ALARP evaluations in relation to severe accidents as follows:

- Methods / technologies for confining a molten core,
- Methods of core or containment cooling,
- Further increasing grace / response times,
- Further capturing / reducing fission products inside containment,
- Design of containment head flange and other systems to protect from containment leakage,
- Method for flammable gas control,
- Containment venting, and
- Additional severe accident management measures.

In each topic, options were considered and the detailed evaluations included assessment of merits and demerits.

25.16.4 Conclusions

As a result of ALARP demonstration, there are no further reasonably practical measures which can be taken in GDA to reduce risks further. Significant recommendations for reasonably practical measures which should be considered for further study at the post GDA phase are representatively presented above. Therefore it is concluded that the UK ABWR design in GDA has no further reasonably practical measures to reduce risks.

25.17 Conclusions

The fuel-scope PSA for UKABWR General Design Assessment (GDA) has been developed through the process of initiating event identification, internal hazard identification/prioritisation and external hazard identification/prioritisation. The following PSA quantification results have been obtained from the UK ABWR PSA.

- Internal Initiating Events at Power,
- Internal Initiating Events during Shutdown,
- Internal Initiating Events for the Spent Fuel Pool,
- Internal Fire for at Power, Shutdown and the Spent Fuel Pool,
- Internal Flooding for at Power, Shutdown and the Spent Fuel Pool,
- Seismic Hazard for at Power, Shutdown and the Spent Fuel Pool,
- Non Reactor Fault including Fuel Route, and
- Other Hazards (Turbine Missile, Tornado Missile and Accidental Aircraft Impact).

Simplified assessments have been performed to assess the risks from the Fuel Routes, non-Reactor Faults, Tornado Missiles, Turbine Missiles and accidental aircraft impact. The evaluation of the Non-Reactor Faults including Fuel Route shows that both BSL and BSO limits are met for all dose levels. These events and Tornado Missiles, Turbine Missiles and Accidental Aircraft Impact were found to have negligible contribution to risk. External flooding and water-based biological fouling are assessed as a sensitivity analysis. Further, potential design enhancements are discussed by using risk insights from these hazards. These enhancements can be used as design options when these hazards are assessed using detailed hazard assessments.

Initial assessments have been performed for the Level 1 and Level 2 Internal Events analyses. Before the preparation for the development of Level 3 PSA, Hitachi-GE updated all of the Internal Events PSAs as well as the detailed Level 1 and Level 2 Hazards analyses for Fire, Flooding, and Seismic. The core damage frequencies and large release frequencies were used as surrogates for Level 3 analysis. Core damage frequency represents an upper bound for significant fission product releases from the fuel; therefore, the core damage frequency bounds the potential for exceeding individual risk. Large Release generally represents a conservative surrogate for the Target 9 Societal Risk because not all sequences resulting in Large Release exceed the Societal Risk Limits.

For Internal Events, the UK ABWR individual risk meets the individual risk criterion defined in the UK ABWR Nuclear Safety and Environmental Design Principles (NSEDP), which are equivalent to Basic Safety Objective of Numerical Target 7 and 8 as defined in ONR's Safety Assessment Principles. The analysis demonstrates that the UK ABWR has met the Basic Safety Levels (BSL) for Internal Events by a considerable margin. The Basic Safety Objectives (BSOs) for Targets 7 and 8 are also met by wide margins while the Target 9 BSO is currently exceeded slightly. It is estimated that the societal risk of the UK ABWR will meet the criterion in NSEDP (equivalent to BSO of Target 9) by crediting some offsite emergency actions.

For internal fires, internal floods and earthquakes, the hazards PSA evaluations have been limited by the available design details. These PSA models have been developed with the latest available design information and will be utilised for design development with coordination between the PSA and Engineering teams. Further evaluations of fire, flood and seismic challenges will be undertaken for ALARP assessment as the UK ABWR design proceeds through detailed design to the post GDA phase (e.g. when procurement information and construction information become available).

Seismic PSA results in GDA are based on generic system and component fragilities. Experiences with plant-specific evaluations have shown that high seismic capacities are achievable. Actual mounting details developed for the plant-specific design are critical to determining plant-specific fragilities. Throughout the detailed design process associated with the post GDA phase, the fragilities will be re-evaluated to ensure that the seismic risks are ALARP.

In conclusion, the internal events PSA results currently demonstrate that the basic design and design features of the UK ABWR are ALARP. When the internal events PSA analyses are enhanced, the risks will likely be reduced. Throughout the progression of the project, the internal events and the all hazards PSA models will be further developed with additional design details. It is expected with reasonable assurance that the detailed design of the UK ABWR will be proven to be truly ALARP.

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Appendix A: Document Map

The following figure shows the PSA document structure.