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UK ABWR Generic Design Assessment

Generic PCSR Chapter 26 : Beyond Design Basis and Severe Accident Analysis

Hitachi-GE Nuclear Energy, Ltd.

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Executive Summary

This analysis chapter extends the UK ABWR Design Basis Analysis safety case that is presented in Chapter 24, by summarising the results of additional fault analysis; specifically Beyond Design Basis (BDB) Analysis and Severe Accident (SA) Analysis. A generic definition of beyond design basis faults is presented in PCSR Chapter 5.

BDB Analysis identifies and analyses credible fault sequences that have lower frequencies than those included in the Design Basis, typically as the result of assuming multiple failures or common cause failures of protection systems following an initiating event. Such fault sequences could potentially result in significant radiological consequences. The aim of the BDB Analysis is to show that such low frequency fault sequences do not contribute disproportionally to nuclear risk i.e. that there is no sudden increase in the severity of the radiological consequences as the Design Basis boundary is crossed. In practice, the analysis presented in this chapter shows that no BDB faults in the GDA fault schedule lead to melting or considerable damage of the core, so that no significant environmental release of any radioactive material occurs.

The SA analysis considers faults at power, faults at shutdown, and faults in the spent fuel pool. The purposes of this SA analysis are to understand the severe accident progression and phenomena and show the effectiveness of countermeasures. The countermeasures include severe accident mitigation systems, described in PCSR Chapter 16, and accident management strategies. The results of the analysis evaluate the CLIFF EDGE effect in terms of radiological consequences when crossing the BDB/SA boundary, and assist in the development of the emergency plans described in PCSR Chapter 22. This analysis also provides source terms for Level 3 Probabilistic Safety Assessment, which considers risks to the public from off-site releases.

The various computer codes used in the BDB and SA analyses are discussed, showing their relevance to specific types of event. The methodology, of necessity, also makes a number of assumptions, which are specified in this chapter. All of the assumptions made are fully consistent with the design information and safety claims for frontline and support systems that are made in the PCSR systems chapters.

The UK ABWR has been designed to minimise the fission product release to the environment in severe accidents. It has a variety of engineered safety features, strategies and procedures for responding to design basis accidents, beyond design basis accidents and severe accidents. These have been assessed against the lessons learnt from the Fukushima accident. It has been confirmed that the UK ABWR has appropriate features already included in the current design, or under design development, to address recommendations made in the UK and worldwide based on the Fukushima lessons learnt. These include the additional risk reduction measure of provision of a Backup Building in the UK ABWR design, which has additional safety features to support core damage prevention and mitigation.

This chapter demonstrates that the risks associated with Beyond Design Basis Events and Severe Accident mitigation for the UK ABWR are capable of being reduced to levels that are As Low As Reasonably Practicable (ALARP). It is acknowledged that further work will be required post-GDA to develop the design and fully incorporate site specific aspects. This work will be the responsibility of any future licensee.

26.1 Introduction

Chapter 26 extends the Design Basis Analysis (DBA) of Chapter 24 to events with lower initiating event or fault sequence frequencies (Beyond Design Basis Accidents (BDBA)) and faults with additional failures leading to high consequences (Severe Accidents (SA)), albeit at very low frequencies.

26.1.1 Background

Chapter 24 of this PCSR describes the assessment of Design Basis (DB) faults, that is, faults with an initiating event frequency greater than 10^{-5}/year or a fault sequence frequency greater than 10^{-7}/year and consequences above Nuclear Safety and Environmental Design Principles (NSEDPs) Basic Safety Limit (BSL) [Ref-47].

Beyond Design Basis Analysis (BDBA) identifies and analyses events that have a lower frequency than DB accidents, usually as the result of multiple failures or common cause failures of protection systems, and with potentially higher consequences, to demonstrate that consequences do not suddenly increase as the DB threshold is passed (so called "CLIFF EDGE" effects). The assessment addresses the adequacy of safety provisions, defence in depth and diversity of protection.

Severe accidents are defined as those fault sequences that could lead either to consequences exceeding the highest off-site radiological doses given in the BSL of the NSEDPs (SP14.2.1), or to an unintended relocation of a substantial quantity of radioactive material within the facility which places a significant demand on any remaining physical barriers.

26.1.2 Document Structure

The chapter is divided into a number of sections:

- Section 26.2 Purpose and Scope,
- Section 26.3 Beyond Design Basis Analysis,
- Section 26.4 Severe Accident Analysis,
- Section 26.5 Fukushima Accident Learning and Stress Test Assessment,
- Section 26.6 Assumptions , Limits and Conditions for Operation,
- Section 26.7 Summary of ALARP Justification, and
- Section 26.8 Conclusions.

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 5: General Design Aspects
  - Definition of Design Basis Faults and Beyond Design Basis Faults including severe accident is summarised in Section 5.5.
  - Categorisation of safety functions and classification of structures, Systems and Components (SSCs) is summarised in Section 5.6.
- Chapter 6: External Hazards
  - Individual external hazards and combined external hazards are identified in Sections 6.3 and 6.4.
  - The Relationship between safety functions and external hazards is summarised in Section 6.5.
  - The information of these sections is used as an input of external hazards to Fault Schedule and PSA (Probabilistic Safety Analysis), which identify accident scenario for Beyond Design Basis Analysis and Severe Accident Analysis.
  - Margin evaluation for Beyond Design Basis Events is summarised in Section 6.7.
- Chapter 7: Internal Hazards
  - Considerations for internal hazards such as internal fire and explosions, internal flooding, pipe whip and jet impact, dropped and collapsed load, internal missiles, internal blast, Electromagnetic Interference(EMI)/Radio Frequency Interference (RFI), Miscellaneous internal hazards, primary containment vessel, main control room, main steam tunnel room, turbine disintegration, and internal combined hazards, are summarised from Sections 7.4 to 7.16.
  - The information of these sections is used as an input of internal hazards to Fault Schedule and Probabilistic Safety Analysis (PSA), which identify accident scenario for Beyond Design Basis Analysis and Severe Accident Analysis.
- Chapter 10: Civil Works and Structures
  - Considerations of civil structures for Beyond Design Basis Faults and Severe Accident are summarised in Section 10.6.
  - The summary of the ultimate capacity of the Reinforced Communication Cabinet (RCCV) containment is provided in Section 10.6.4.
- Chapter 11: Reactor Core
  - A brief overview of the designs of the ABWR reactor system, core and fuel is summarised in Section 11.3.
  - Design description of reactor core is summarised in Section 11.5.
  - The information of these sections is used as an input of reactor core systems to Beyond Design Basis Analysis and Severe Accident Analysis.
- Chapter 12: Reactor Coolant Systems, Reactivity Control Systems and Associated Systems
  - Reactor coolant systems and associated systems such as RHR (Residual Heat Removal) are summarised in Section 12.3.
  - Reactivity control systems such as Control Rod Drive (CRD) and Standby Liquid Control System (SLC) are summarised in Section 12.4.

  - The information of these sections is used as an input of reactor coolant systems and reactivity control systems to Beyond Design Basis Analysis and Severe Accident Analysis.
- Chapter 13: Engineered Safety Features
  - Containment system is summarised in Section 13.3.
  - Emergency Core Cooling System is summarised in Section 13.4.
  - The information of these sections is used as an input of the containment and the Emergency Core Cooling Systems (ECCSs) to Beyond Design Basis Analysis and Severe Accident Analysis.
- Chapter 14:Control and Instrumentation
  - The overall architecture of the C&I systems for the UK ABWR is summarised in Section 14.5.
  - The overview of the role and safety justification of each specific C&I system including Severe Accident C&I system is summarised in Section 14.6.
  - The information of these sections is used as an input of the C&I systems to Beyond Design Basis Analysis and Severe Accident Analysis.
- Chapter 15: Electrical Power Supplies
  - The key features of the Electrical Power System (EPS) architecture and power supplies are summarised in Section 15.4.
  - Electrical equipment and systems such as Emergency Diesel Generators, Backup Building (B/B) Generator, Diverse Additional Generator, and Power Trucks are summarised in Section 15.5.
  - The information of these sections is used as an input of the electrical systems to Beyond Design Basis Analysis and Severe Accident Analysis.
- Chapter 16: Auxiliary Systems
  - Water systems such as ultimate heat sink, Reactor building Cooling Water system (RCW), and Reactor building Service Water system (RSW) are summarised in Section 16.3.
  - Process auxiliary systems such as high pressure nitrogen gas supply system are summarised in Section 16.4.
  - Heating Ventilating and Air Conditioning system (HVAC) is summarised in Section 16.5.
  - Other auxiliary systems such as fire protection systems and emergency power supply systems are summarised in Section 16.6.
  - Severe Accident Mechanical systems are summarised in Section 16.7.
  - The information of these sections is used as an input of the mechanical systems to Beyond Design Basis Analysis and Severe Accident Analysis.
- Chapter 19: Fuel Storage and Handling
  - Fuel preparation machine, fuel handling machine, and reactor building overhead crane are summarised in Sections 19.5, 19.6 and 19.7.
  - Spent fuel storage facility, fuel pool cooling, clean-up and makeup systems, and spent fuel export systems are summarised in Sections 19.8, 19.9, and 19.10.

  - The information of these sections is used as an input of the mechanical systems in the Spent Fuel Pool (SFP) to Beyond Design Basis Analysis and Severe Accident Analysis.
- Chapter 20: Radiation Protection
  - Post accident accessibility, i.e. dose assessment for the workers mitigating fault / accident, is summarised in Section 20.9.
- Chapter 22: Emergency Preparedness
  - This chapter gives an input of safety and emergency systems for Section 22.4.
  - This chapter give an input of Fukushima lessons learnt for Section 22.5.
  - This chapter gives an input of accident management for Section 22.7.
- Chapter 24: Design Basis Analysis – defines DB starting point for BDBA
  - The transient analyses performed for the DB assessment provide initial conditions for the Beyond Design Basis Assessment. Fault Identification and Grouping and Fault Schedule are summarised in Section 24.4.
- Chapter 25: PSA
  - Level 1 PSA of internal event at power, which is described in Section 25.4, identifies severe accident scenario.
  - Severe accident analysis gives an input of qualification of the containment event tree for Level 2 PSA of internal event at power, which is described in Section 25.5.
  - Severe accident analysis gives an input of the source term for Level 2 PSA of internal event at power, which is described in Section 25.5.
  - Shutdown PSA, which is described in Section 25.8, identifies severe accident scenario during shutdown modes. Severe accident analysis gives an input of the source term for the shutdown PSA.
  - SFP PSA, which is described in Section 25.9, identifies severe accident scenario in the SFP. Severe accident analysis gives an input of the source term for the SFP PSA.
  - Internal hazard PSA, which is described in Section 25.10, identifies severe accident scenario due to internal hazards. Severe accident analysis gives an input of qualification of the containment event tree and the source term for the internal hazard PSA.
  - External hazard PSA, which is described in Section 25.11, identifies severe accident scenario due to external hazards. Severe accident analysis gives an input of qualification of the containment event tree and the source term for the external hazard PSA.
  - PSA insight and ALARP assessment in PSA, which are described in Sections 25.15 and 25.16, give an input for ALARP assessment for Beyond Design Basis Accident and Severe Accident.
- Chapter 27: Human Factors (HF)
  - Operator actions in Beyond Design Based Accident and Severe Accident are supported by HF analysis, which is described in Section 27.5.
- Chapter 28: ALARP Evaluation

- This chapter gives an input of ALARP insights from Beyond-Design Basis Accident and Severe Accident, which is described in Section 28.6.3.
- This chapter gives an input of Design decisions arising from insights from the BDBA and Severe Accident Analysis (SAA), which is described in Section 28.7.3.

The relationship between the SAA presented in this chapter and the Safety Functional Claims (SFCs) as well as the Safety Properties Claims (SPCs) used elsewhere in the PCSR are described in Appendices A and B, respectively. The chapter is supported by a number of Topic Reports (TRs) on Fault Assessment, BDB Assessment, and SAA. These TRs are shown in the document map along with other relevant PCSR chapters in Appendix C.

26.2 Purpose and Scope

26.2.1 Purpose

The purpose of this chapter is to extend the DB analysis of Chapter 24 beyond the design basis and to provide assessments of accidents with potentially severe consequences in order to support the PSA in Chapter 25, the accident management aspects of Emergency Preparedness in Chapter 22 and the ALARP Evaluation in Chapter 28.

The BDB assessment identifies and groups BDB faults, that is, faults with frequency outside the range of DB faults (as defined in Chapter 5), and provides analysis to demonstrate that there are no "CLIFF EDGE" effects near the cut-off frequency and that the risks are ALARP. DB faults do not lead to core damage or to releases of radiation leading to doses to public or workforce which exceeds the established acceptance criteria for the DBA (as defined in Chapter 24). The BDB analysis shows that such faults do not lead to melting or considerable damage of the core so that no significant environmental release of any radioactive material occurs. This is achieved through demonstration of the adequacy of safety provisions based on defence in depth and diversity in the design.

The Severe Accident analysis has been performed to analyse hypothetical fault sequences that result in significant radiological consequences. The purpose of these severe accident analyses is:

- To understand the severe accident phenomena and progression including fission product release and transport behaviour.
- To determine the response times for accident management such as equipment recovery and operator action.
- To evaluate engineered features, strategies, and procedures for mitigating a severe accident.
- To demonstrate the effectiveness of engineered features, strategies, and procedures.
- To determine the magnitude and characteristics of the potential radiological consequences, including societal risk.
- To demonstrate that there is no sudden escalation of consequences just beyond the design basis.
- To support the Level 2 PSA quantification

26.2.2 Scope

Using the Fault Schedule in Chapter 24, the BDB assessment identifies and groups BDB faults and provides acceptance criteria for their analysis. The chapter continues by discussing analysis codes and analysis conditions and provides the results of the corresponding BDB analysis to demonstrate that there are no "CLIFF EDGE" effects as conditions move beyond what is considered in the DB analysis.

The Severe Accident assessment provides a safety case strategy for severe accidents and their management based on the following aspects:

- Providing understanding of the physical progression of Severe Accidents,
- Developing Strategies, Measures and Procedures for Severe Accidents,
- Examining the response to the Fukushima Accident and the Stress Test Report,
- Describing SA Analysis Codes and providing SA analysis,
- Evaluating the probability of PCV (Primary Containment Vessel) Failure,
- Analysing Severe Accident Analysis at Power,
- Analysing Severe Accident Analysis during Shutdown, and
- Severe Accident Analysis in the Spent Fuel Pool (SFP).

The Severe Accident assessment provides specific input to Chapter 22 on Emergency Preparedness by providing information on available timescales and the strategies and procedures that would be used to manage severe accidents. It also provides specific inputs to Chapter 25 on PSA in that it provides understanding of severe accident phenomena and progression, including fission product release and transport behaviour and provides information to determine the magnitude and characteristics of the predicted source term as an input to the evaluation of potential radiological consequences, including societal risk. For links to the Generic Environmental Permit (GEP) and the Conceptual Security Arrangements (CSA) documentation, please refer to PCSR Chapter 1. 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.

26.3 Beyond Design Basis Analysis

Beyond Design Basis (BDB) faults are those events which have frequency lower than the lowest Design Basis faults or fault sequences. The requirement for BDB faults is that there should not be a sudden increase in mitigated consequences as the frequency of events goes below the lowest DB frequency (no "CLIFF EDGE") and that the doses should be ALARP.

26.3.1 Safety Case Strategy for Beyond Design Basis Faults

In the Design Basis Safety Case presented in Chapter 24, it is demonstrated that there is no consequential fuel damage and no consequential failure of the reactor coolant pressure boundary or primary containment for any DB fault. This means that onsite and off-site doses are limited to those arising from the release of primary coolant from DB faults. Limits and Conditions for Operation on primary coolant activity levels ensure that these doses are significantly below the BSL for the corresponding NSEDP DB dose targets.

The conclusion that there is no consequential fuel damage and no consequential failure of the reactor coolant pressure boundary or primary containment for any DB fault is ensured by showing that corresponding Acceptance Criteria relating to fuel integrity, reactor coolant pressure boundary integrity and primary containment integrity are met. The basic strategy for demonstrating a safety case for BDB faults is to show that the same Acceptance Criteria are met as for infrequent DB faults. This ensures that the mitigated consequences of BDB fault are no worse than those for infrequent DB faults and hence no "CLIFF EDGE".

This strategy has a number of steps:

- Identification and grouping of BDB faults,
- Identification of relevant Acceptance Criteria from DB fault studies,
- Identification of SSCs that provide protection against BDB faults, and
- Transient analysis of the identified events to show Acceptance Criteria are met.

(a) Identification and Grouping of BDB faults

The identification of BDB faults starts with the same initiating faults as identified for DB fault studies and then looks at additional protection system common cause failures that would take the event outside the Design Basis by up to an order of magnitude. Grouping of BDB faults is simpler than for DB faults as only three basic groups are identified:

- BDB faults at power,
- BDB faults in shutdown modes, and
- BDB faults in SFP and fuel route.

(b) Identification of relevant Acceptance Criteria from DB fault studies

There are different Acceptance Criteria in the DB fault analysis for frequent and infrequent faults. BDB faults are even less frequent than infrequent DB faults. Therefore it is appropriate to adopt the same Acceptance Criteria of the fuel and the reactor coolant pressure boundary as infrequent faults for BDB faults. For the Acceptance Criteria of the primary containment, the limits and conditions are known and therefore they are adopted for BDB faults. For faults in the SFP and fuel route, the Acceptance Criteria mainly relate to water levels to ensure fuel is always cooled. For the Acceptance Criteria of dose, the different Acceptance Criteria are adopted for BDB faults from DB faults, which are the same as PSA because the nominal conditions are assumed in the BDB analysis. Appendix A gives the link between the Acceptance Criteria and SFCs in Chapters 11, 12 and 13 of this PCSR.

(c) Identification of SSCs that provide protection against BDB faults

It is assumed for BDB faults that all Class 1 SSCs are unavailable in a High Level Safety Function (HLSF), usually because it is the common cause failure of a Class 1 SSC leading to the loss of the Class 1 provision of a HLSF that leads to a DB fault becoming a BDB fault. This is a conservative assumption because it may well be that Class 1 provision of other HLSFs is still available. The available Class 2 SSCs are identified along with the HLSFs they provide. Appendix A also lists the specific SFCs associated with the SSCs.

(d) Transient Analysis

The transient analysis uses some of the same computer codes as the DB analysis but uses the nominal accident initial conditions. The provision of HLSFs is different for BDB faults. The results shown in this section demonstrate that the corresponding Acceptance Criteria are met for BDB faults and that there is no “CLIFF EDGE”.

26.3.2 Identification and Grouping of Beyond Design Basis Faults

Based on the NSEDPs [Ref-47] and PCSR Chapter 5, Section 5.5, a fault which satisfies one of the following two conditions is defined as a BDB fault.

- Frequency of initiating fault < 10^{-5} /year, and
- Frequency of fault sequence < 10^{-7} /year.

In the following sections, a fault is identified and grouped for the beyond design basis fault in power operation and shutdown mode [Ref-40] [Ref-41] and for the beyond design basis fault for SFP and fuel route [Ref-42]. These are the same operating modes used in the design basis analysis in Chapter 24.

26.3.2.1 Beyond Design Basis Fault at Power

To identify a beyond design basis fault, combination of an initiating fault and Common Cause Failure (CCF) of HLSFs is assumed. The following method is applied to the UK ABWR, that is, firstly initiating faults are identified, secondly CCF of HLSFs is identified, and finally frequency of fault sequence is estimated and faults are grouped.

(1) Identification of initiating faults

The same initiating faults as used in the DB analysis in Chapter 24, Section 24.4.5 are considered. The faults are grouped as transient and LOCA (Loss Of Coolant Accident) based on the frequency but LOOP (Loss Of Off-site Power) is not included in transient because the frequency depends on the duration of LOOP as well as break size of LOCA. The initiating faults are:

- Non LOOP transient (Fault Schedule Ref: 1 to 4, and 6),
- Short-Term LOOP (Fault Schedule Ref: 5.1),
- Medium-Term LOOP (Fault Schedule Ref: 5.2),
- Long-Term LOOP (Fault Schedule Ref: 5.3),
- Small Break LOCA (Fault Schedule Ref: 7),
- Medium Break LOCA (Fault Schedule Ref: 8), and
- Large Break LOCA (Fault Schedule Ref: 9).

(2) Identification of CCF of HLSFs

CCF in systems providing HLSFs is considered. HLSF is the function to mitigate the fault progression and the HLSFs are “Reactivity Control”, “Fuel Cooling”, “Long-term Heat Removal”, and “Confinement and containment” in accordance with the fault schedule.

(3) Simple estimation of frequency of fault sequence and grouping of faults

Initiating faults are separated into following three groups, Transient without LOOP, LOOP, and LOCA. In addition, LOOP is separated into three types based on the duration, and LOCA is separated into three types based on the cooling system break size.

For faults sequences, CCF of the primary safety system is considered and then CCF of the secondary safety system is considered in each HLSF. When CCF of the secondary system is considered, the frequencies of CCF of the primary system and of the secondary system are cumulated to show the total frequency.

As the result, the following faults to be analysed as BDB faults are identified. It is noted that “Rupture of one outboard check valve with failure of one inboard check valve at feedwater line” is identified to decide the structural integrity classification of outboard isolation valves.

(1) Long-term LOOP with failure of scram,

(2) Medium Break LOCA with failure of scram,

(3) Non-LOOP transient with failure of scram and ARI (Alternative Rod Insertion),

(4) Medium-term LOOP with failure of scram and ARI,

(5) Long-term LOOP with CCF of EDGs (Emergency Diesel Generators),

(6) Medium Break LOCA with CCF of EDGs,

(7) Small Break LOCA with CCF of EDGs and failure of RCIC (Reactor Core Isolation Cooling System),

(8) Medium-term LOOP with CCF of EDGs and failure of RCIC,

(9) Non-LOOP transient with CCF of RHR and failure of containment venting,

(10) Short-term LOOP with CCF of RHR and failure of containment venting,

(11) Rupture of one outboard MSIV (Main Steam Isolation Valve) with failure of one inboard MSIV, and

(12) Rupture of one outboard check valve with failure of one inboard check valve at feedwater line.

26.3.2.2 Beyond Design Basis Fault in Shutdown Mode

To identify a beyond design basis fault, combination of an initiating fault and CCF of HLSFs is assumed. The following method is applied to the UK ABWR, that is, firstly initiating faults are identified, secondly CCF of HLSFs is identified, and finally frequency of fault sequence is estimated and faults are grouped.

(1) Identification of initiating faults

The same initiating faults as DB faults in Chapter 24, Section 24.4.5 are considered, which are as follows. They are considered in Operating States C-1 to C-5, which are shown in Chapter 24, Section 24.10.2.

- Loss of Operating RHR with the same division ECCS (Fault Schedule Ref: 13.3),
- Loss of Operating RHR due to CCF of Class1 controller (Fault Schedule Ref: 13.4),
- Short-Term LOOP (Fault Schedule Ref: 13.5.1),
- Medium-Term LOOP (Fault Schedule Ref:13.5.2),
- Long-Term LOOP (Fault Schedule Ref:13.5.3),
- Leakage due to maintenance (Fault Schedule Ref: 13.13, 13.14, and 13.15),
- LOCA (Fault Schedule Ref: 13.7 to 13.12 and 13.16 to 13.18),
- CCF of C&I Systems (Fault Schedule Ref: 11.4.2, 11.6, and 11.7),
- CCF of Electrical Power Supply Systems (Fault Schedule Ref: 11.8.2), and
- CCF of Essential Services and Support Systems (Fault Schedule Ref: 11.10.2, 11.11.2, and 11.12.2).

(2) Identification of CCF of HLSFs

CCF in systems providing HLSFs is considered. HLSF is the function to mitigate the fault progression, and basically HLSF claimed in the fault schedule in shutdown modes are “Fuel Cooling” and “Long-term Heat Removal”. HLSFs “Reactivity Control” is not considered because reactor shutdown state is achieved in advance. “Confinement” is also not considered because confinement function is not claimed in DBA with an exception that HLSF “Confinement” is considered in LOCAs outside PCV to mitigate draindown of the reactor and internal flooding. Meanwhile, it is remarked that frequency of fault sequence are estimated depending on the number of operating and standby systems.

(3) Simple estimation of frequency of fault sequence and grouping of faults

Simple estimation of frequency of fault sequence for the all initiating faults is carried out for the each Operating State.

For faults sequences, CCF of the primary system is considered and then CCF of the secondary system is considered in each HLSF. When CCF of the secondary system is considered, the frequencies of CCF of the primary system and of the secondary system are cumulated as the total frequency. It is noted that the frequencies of CCFs are determined by number of trains in operation or standby. The number of available trains are depend on unavailability of system due to maintenance based on typical outage schedule in shutdown modes, and unavailability of system due to consequential effect of initiating event to the safety divisions. In addition, LOCA (mechanical) below TAF assuming guillotine break is also considered in beyond design basis analysis.

As the result, the following faults to be analysed as BDB faults are identified.

(1) Loss of Operating RHR with Loss of all ECCS and Failure of FLSS (Operating state C-1, C-, C-3-1 and C-5),

(2) Loss of Operating RHR with Loss of all ECCS and Failure of FLSS and FLSR (Operating state C-3-2, C-3-3 and C-4),

(3) Short term Loss of Offsite Power (LOOP)with CCF of EDGs and BBGs (Operating state C-1 and C-2) ,

(4) Medium term Loss of Offsite Power (LOOP) with CCF of EDGs and BBGs (Operating state C-3 and C-4) ,

(5) Long term Loss of Offsite Power with CCF of EDGs (Operating state C-1 to C-5) ,

(6) LOCA at Feedwater line inside PCV with Loss of all ECCS (Operating state C-3 and C-4),

(7) LOCA (mechanical) below TAF (Operating state C-1, C-2 and C-5),

(8) LOCA (mechanical) below TAF with Loss of all ECCS (Operating state C-3 and C-4), and

(9) Inadvertent start-up of A2 injection systems with Loss of all ECCS and failure of FLSS and FLSR (Operating state C-3-1).

Note that the fault sequences for each initiating event in shutdown modes could be plural because plant status such as RPV/PCV head open/close, SFP gate open/close or initial reactor water level, and frequency of the fault sequences are different since number of operating and standby system are different for each operating states.

26.3.2.3 Beyond Design Basis Fault for SFP and Fuel Route

To identify a beyond design basis fault, combination of an initiating fault and CCF of HLSFs is assumed. The following method is applied to the UK ABWR, that is, firstly initiating faults are identified, secondly CCF of HLSFs is identified, and finally frequency of fault sequence is estimated and faults are grouped.

(1) Identification of initiating faults

The same initiating faults as DB faults for SFP and Fuel Route in Chapter 24, Section 24.4.5 are considered, which are as follows:

- SFP faults
    - Medium-Term LOOP (Fault Schedule Ref: 14.2.2),
    - Long-Term LOOP (Fault Schedule Ref: 14.2.3),
    - Loss of all FPC pumps (Fault Schedule Ref: 14.1),
    - Small leak of SFP (Fault Schedule Ref: 14.4.1), and
    - FPC line break (Fault Schedule Ref: 14.4.2).

- Fuel Route faults
    - Over-raise of irradiated fuel by FHM main hoist (Fault Schedule Ref: 14.9), and
    - Drop of Cask with Loaded canister with water into the SFP (Fault Schedule Ref: 14.8).

In addition, dropping of heavy equipment is considered as a BDB fault.

(2) Identification of CCF of HLSFs

- SFP faults

CCF in systems providing HLSFs is considered. HLSF is the function to mitigate the fault progression and HLSFs for SFP faults are “Reactivity Control”, “Fuel Cooling” and “Confinement/Containment of Radioactive Materials” in accordance with the fault schedule. For “Reactivity Control”, it is noted that sub-criticality is maintained by SFP rack which is a passive component in both a normal and a fault condition. Thus, fault escalation due to failure of class 1 reactivity control function is not considered as a BDB fault. In addition, “Confinement/Containment of Radioactive Materials” function is not treated as DBA claim for SFP faults and therefore there is no escalation of consequence with regard to “Confinement” function just beyond the design basis for this fault group.

- Fuel Route faults

HLSF is the function to mitigate the fault progression and the HLSF for Fuel Route faults such as drop of fuel /heavy equipment and the load over-raise event is "Confinement/Containment of Radioactive Materials" including radiation shielding in accordance with the fault schedule.

For drop of fuel /heavy equipment except for cask events, unmitigated potential radiological consequences to the public and the worker do not exceed the BSL with the frequency of infrequent initiating fault given in the NSEDPs (SP14.2.1), and that there is no escalation of consequence just beyond the design basis for this fault group. On the other hand, the preliminary unmitigated radiation dose is conservatively calculated following an unmitigated cask drop fault onto the truck bay, and concludes that the unmitigated consequence could exceed the BSL given in the NSEDPs (SP14.2.1). Furthermore, over-raise event such as over-raise of irradiated fuel unmitigated potential radiological consequences to the worker are evaluated to be greater than the BSL with the frequency of the frequent initiating fault. Therefore, BDB faults which are relevant to cask drop and over-raise event are investigated in the same way as SFP faults. The systems or components in each HLSF are as follows:

- Containment of Radioactive Materials:
  - Impact limiter, and
  - Canister containment boundary.

- Confinement/Containment of Radioactive Materials (Shield Radiation):
  - Class1 limit switch, and
  - Class2 limit switch.

(3) Simple estimation of frequency of fault sequence and grouping of faults

- SFP faults

Initiating faults are separated into three fault groups and five faults;

- Loss of decay heat removal (Loss of all FPC pumps) (Fault Schedule Ref: 14.1),
- LOOP (Medium-term LOOP and Long-term LOOP) (Fault Schedule Ref: 14.2.2 and 14.2.3), and
- Loss of water inventory (Small leak of SFP and FPC line break) (Fault Schedule Ref: 14.4).

It is noted that short term LOOP is excluded because boiling does not occur in the SFP during short term (2 hours) LOOP even in the maximum heat load condition.

For faults sequences, CCF of the primary system is considered and then CCF of the secondary system is considered in HLSF for fuel cooling. When CCF of the secondary system is considered, the CCF probabilities of the primary system and of the secondary system are cumulated as the total CCF probabilities.

- Over-Raise Events

The following fault group and one fault are considered for identification of BDB faults because their unmitigated radiological consequences could exceed the BSL given in the NSEDPs (SP14.2.1);

- Over-raise of irradiated fuel by FHM main hoist (Fault Schedule Ref: 14.10)

For faults sequences, CCF of the primary system is considered and then CCF of the secondary system is considered in HLSF for shielding radiation to the worker. When CCF of the secondary system is considered, the CCF probabilities of the primary system and of the secondary system are cumulated as the total CCF probabilities.

- Drop of Heavy equipment

As described above, the event caused by multiple failures such as drop of heavy equipment is considered as a BDB fault. The following event is identified as a bounding BDB fault relevant to drop of heavy equipment.

- Drop of Cask with Loaded canister with water into the SFP (Fault Schedule Ref: 14.8)

As a result of identification of BDB faults for SFP and fuel route, the following five faults to be analysed as BDB faults are identified:

(1) Loss of all FPC pumps with failure of FLSS,

(2) Long term SBO,

(3) Small leak of SFP with failure of FPC and FLSS,

(4) Over-raise of irradiated fuel by FHM main hoist with failure of Class 1 limit switch and Class 2 limit switches, and

(5) Drop of Cask with Loaded canister with water into the SFP.

26.3.3 Identification of Relevant Acceptance Criteria for Beyond Design Basis Faults

SP14.3.2 and SP14.3.3 of the NSEDPs are applied to BDB fault [Ref-47]. The NSEDPs define two types of safety level with different numerical targets. These are Basic Safety Levels (BSLs) and Basic Safety Objectives (BSOs).

Acceptance criteria relating directly to meeting NSEDP dose targets (off-site and on-site) for BDB faults are designated as 'AC-Dx'.

To confirm compliance with SP14.3.3 of the NSEDPs, the total predicted frequencies of accidents on an individual facility, which could give doses to a person off the site, shall not exceed the target below. In particular, the following target with a fraction of 10 on frequency is confirmed to each BDB fault safety case.

Effective dose, mSv	**Total predicted frequency per annum**		**Acceptance Dose Criteria for off-site**
	BSL	**BSO**	
0.1 to 1	1	1×10^{-2}	(AC-D7)
1 to 10	1×10^{-1}	1×10^{-3}	(AC-D8)
10 to 100	1×10^{-2}	1×10^{-4}	(AC-D9)
100 to 1,000	1×10^{-3}	1×10^{-5}	(AC-D9)
> 1,000	1×10^{-4}	1×10^{-6}	(AC-D10)

To confirm compliance with SP14.3.2 of the NSEDPs, the predicted frequency of any single accident in the facility, which could give doses to a person on the site, shall not exceed target below.

Effective dose, mSv	**Total predicted frequency per annum**		**Acceptance Dose Criteria for On-site**
	BSL	**BSO**	
2 to 20	1×10^{-1}	1×10^{-3}	(AC-D11)
20 to 200	1×10^{-2}	1×10^{-4}	(AC-D12)
200 to 2,000	1×10^{-3}	1×10^{-5}	(AC-D13)
> 2,000	1×10^{-4}	1×10^{-6}	(AC-D14)

All BDB faults should meet criteria AC-D7 to AC-D14, although application of the ALARP principle leads to much lower doses being achieved in practice. The dose-risk targets for workers and the public are discussed in Chapter 20.

Specific acceptance criteria at power, for shutdown modes and SFP and Fuel Route faults are shown in the following subsections.

26.3.3.1 Beyond Design Basis Faults at Power

For BDB faults, the following five Acceptance Criteria are applied to meet the targets state above. These Acceptance Criteria indicate that the excess embrittlement of fuel cladding is prevented from Chapter 11 Section 11.4.2.1, and the reactor coolant pressure boundary and reactor containment boundary are maintained from Chapter 12, Section 12.3.3.3 and Chapter 10, Section 10.6.4.

AC-F5: The calculated maximum fuel cladding temperature shall not exceed 1,200 °C.

AC-F4 The calculated total oxidation of the fuel cladding shall not exceed 15 percent of the total cladding thickness before oxidation.

AC-R2: Pressure on the reactor coolant pressure boundary shall be maintained below 120 percent of the maximum allowable working pressure.

AC-C3 Pressure on the reactor containment boundary shall be maintained below the limiting pressure.

AC-C4: Temperature on the reactor containment boundary shall be maintained below the limiting temperature.

26.3.3.2 Beyond Design Basis Faults in Shutdown Modes and Beyond Design Basis SFP and Fuel Route Faults

For beyond design basis faults in shutdown modes and for SFP and Fuel Route faults, the following Acceptance Criteria are applied. These criteria aim to maintain the integrity of fuel cladding and not release radioactivity to the environment. These are considered as candidates for BDB analysis:

AC-W1: RPV water level shall be maintained above the TAF of the reactor core during shutdown to prevent the fuel being uncovered and heating up.

AC-W2: SFP water level shall be maintained above the TAF of the spent fuel pool to prevent spent fuel being uncovered and heating up.

26.3.4 Identification of SSCs that provide protection against BDB faults

26.3.4.1 Beyond Design Basis Fault at Power

It is assumed that CCF results in the unavailability of Class 1 provisions of the HLSF that is assumed and Table 26.3-1 lists the diverse Class 2 & 3 systems that are available to provide the HLSFs in these circumstances defined in Chapter 5, Section 5.6, Table 5.6-1.

Table 26.3-1 : Provision of Safety Functions

Reactivity Control – Reactor		**HLSFs provided**
Class 2	• Hardwired Back-up System (HWBS) (see Section 14.6.3) • Standby Liquid Control System (SLC) (see Section 11.5.3, Section12.4.3.2 and Section14.6.3) • Recirculation Pump Trip (RPT) (see Section 14.6.3) • Feed water Stop (see Section 14.6.3) • Alternative Rod Insertion (ARI) (see Section 11.5.2, Section12.4.3.1 and Section 14.6.3)	1-5
Fuel Cooling – Reactor		**HLSFs provided**
Class 2	• Hardwired Back-up System (HWBS) (see Section 14.6.3) • Reactor Depressurisation Control Facility (RDCF) (Alternative SRV) (see Section 16.7.3.3) • Flooding System of Specific Safety Facility (FLSS) (see Section 16.7.3.1)	2-2

Table 26.3-1 : Provision of Safety Functions (Cont'd)

Fuel Cooling – Reactor		**HLSFs provided**
Class 3	• Reactor Depressurization Control Facility (Switching Valve for SRVs) (RDCF) (see Section 16.7.3.3) • Flooding System of Reactor Building (FLSR) (see Section 16.7.3.2)	
Long-term Heat Removal – Reactor		**HLSFs provided**
Class 2	• Hardwired Back-up System (HWBS) (see Section 14.6.3) • Atmospheric Control System (AC) (see Section 13.3.3.4) • Filtered Containment Venting System (FCVS) (see Section 13.3.3.4 and Section 16.7.3.5)	3-2
Class 3	• Alternate Heat Exchange Facility (AHEF) (see Section 16.7.3.6)	
Other Functions – Reactor AC power supply function is required to protect against BDB faults depending on fault progression. UK ABWR has the following safety systems to provide these functions.		**HLSFs provided**
Class 2	• Back-up Building Generator System (BBG) (see Section 15.4.6)	1-5, 2-2, 3-2, 4-7, 5-3
Class 3	• Diverse Additional Generator (DAG) (see Section 15.4.7)	5-10

26.3.4.2 Beyond Design Basis Fault in Shutdown Modes

It is assumed that CCF results in the unavailability of provisions of the HLSF that is assumed. Table 26.3-2 lists the safety systems that are available to provide the HLSFs in these circumstances defined in PCSR Chapter 5, Section 5.6, Table 5.6-1.

Table 26.3-2 : Provision of Safety Functions against BDBF in shutdown modes

Fuel Cooing and Long Term Heat Removal - Shutdown Mode		**HLSF provided**
Class 1	• High Pressure Core Flooder system (HPCF) (see Section 13.4.1) • Low Pressure Core Flooder System (LPFL) (see Section 13.4.1) • Residual Heat Removal System (RHR) (see Section 12.3.5.4) • Safety Relief Valve (SRV) –Manual depressurisation– (see Section 12.3.5.2)	2-1, 3-1
Class 2	• Flooding System of Specific Safety Facility (FLSS) (see Section 16.7.3.1) • Reactor Depressurization Control Facility (RDCF) (Alternative SRV) (see Section 16.7.3.3) • Atmospheric Control System (AC) (see Section 13.3.3.4) • Filtered Containment Venting System (FCVS) (see Section 13.3.3.4 and Section 16.7.3.5)	2-2, 2-5, 3-2

Table 26.3-2 : Provision of Safety Functions against BDBF in shutdown modes (Cont'd)

Fuel Cooing and Long Term Heat Removal - Shutdown Mode		HLSF provided
Class 3	• Flooding System of Reactor Building (FLSR) (see Section 16.7.3.2) • Fire Protection System (see Section 16.6.1) • Reactor Depressurization Control Facility (Switching Valve for SRVs) (RDCF) (see subsection 16.7.3.3) • Make-up Water Condensate System (MUWC) (see Section 16.3.4) • Suppression Pool Clean-up System (SPCU) (see Section 16.6.3)	2-2, 2-3, 2-5
Other Functions		**HLSF provided**
Class 2	• Hardwired Back-up System (HWBS) (see Section 14.6.3) • Back-up Building Generator System (BBG) (see Section 15.4.6)	2-2, 2-5, 3-2, 5-3, 5-18
Class 3	• Diverse Additional Generator (DAG) (see Section 15.4.7)	5-10

26.3.4.3 Beyond Design Basis Fault for SFP and Fuel Route

It is assumed that CCF results in the unavailability of Class 1 provisions of the HLSF that is assumed and Table 26.3-3 lists the diverse Class 2 & 3 systems that are available to provide the HLSFs in these circumstances defined in PCSR Capter5, Section 5.6, Table 5.6-1.

Table 26.3-3 : Provision of Safety Functions

Fuel Cooling – Spent Fuel Pool		**HLSFs provided**
Class 2	• Flooding System of Specific Safety Facility (FLSS) (see Section 16.7.3.1)	2-5
Class 3	• Flooding System of Reactor Building (FLSR) (see Section 16.7.3.2) • Fire Protection System (see Section 16.6.1) • Make-up Water Condensate System (MUWC) (see Section 16.3.4) • Suppression Pool Clean-up System (SPCU) (see Section 16.6.3)	
Confinement /Containment of radioactive materials		**HLSFs provided**
Class 2	• Secondary Containment (see Section 13.3.4) • Reactor Area (R/A) Heating Ventilating and Air Conditioning System (HVAC) isolation damper (see Section 16.5) • Standby Gas Treatment System (SGTS) (see Section 13.3.4.2)	4-7
Other Functions		**HLSF provided**
Class 2	• Hardwired Back-up System (HWBS) (see Section 14.6.3) • Back-up Building Generator System (BBG) (see Section 15.4.6)	2-5, 5-2, 5-3. 5-18

26.3.5 Transient Analysis

26.3.5.1 Analysis Code

Table 26.3-4 lists the computer codes used for BDB analysis at power.

Table 26.3-4 : Computer Codes for Beyond Design Basis Analysis at Power

Codes	Applications (phenomena/transients)
SAFER or TRACG	Reactor thermal hydraulic transient
	Water level inside reactor vessel
	Core heat up
MAAP or SHEX	Containment pressure and temperature
RADTRAD	Dose evaluation

Table 26.3-5 summarises calculation methods which are used for BDB analysis in shutdown modes and for SFP and Fuel Route.

Table 26.3-5 : Calculation Method for Beyond Design Basis Analysis in Shutdown Modes and for SFP and fuel route

Codes/tools	Applications (phenomena/transients)
Spreadsheet	Water level and generated steam for shutdown mode
SHEX	Pressure (RPV head closed)
MCNP	• Dose evaluation by direct radiation of a fixed radiation source • On-site dose calculation
RADTRAD	• Dose evaluation of breath and radiation cloud around human, with diffusion of radiation • Off-site dose calculation
ORIGEN2	Calculation of amount of radionuclide inside core considering fuel exposure and composition, which is used as the source term for dose evaluation.

It is noted that the same codes as the DB analysis in PCSR Section 24.5.3 are used.

26.3.5.2 Analysis Assumptions

Analysis conditions are shown in Table 26.3-6 for all faults, Table 26.3-7 at Power, Table 26.3-8 in Shutdown Modes, and Table 26.3-9 in SFP.

Table 26.3-6 Analysis Conditions of Injection Systems

Items		Conditions	Remarks
RCIC			
	Flow rate	182 m^3/h	At 8.115 to 1.034 MPa[dif]
	Actuation signal	Low water level (Level 1.5)	
HPCF			
	Flow rate (1 train)	727 m^3/h	At 0.689 MPa[dif]
	Actuation signal	Low water level (Level 1.5)	
LPFL			
	Flow rate (1 train)	954 m^3/h	At 0.276 MPa[dif]
	Actuation signal	Low water level (Level 1)	
ADS			
	Actuation signal	Low water level (Level 1)	
FLSS for Core			
	Flow rate (1 train)	0 m^3/h 660 m^3/h 1100 m^3/h	At 1.6 MPa[dif] At 1.0 MPa[dif] At 0.0 MPa[dif]
	Actuation signal	Low water level (Level 1)	
	Delay timer	10 min	
RDCF			
	Actuation signal	Low water level (Level 1)	
	Delay timer	10 min	
FLSR			
	Flow rate (1 train)	90 m^3/h 120 m^3/h	For RPV injection For SFP injection (At 0.6 MPa[dif])

Table 26.3-7 Analysis Conditions at Power

Items	Conditions	Remarks
Reactor thermal power	3926 MW	Rated thermal power
Core flow rate	52.2×10^3 t/h	Rated flow rate
Dome pressure	7.07 MPa[gauge]	Rated pressure
Reactor initial water level	Normal water level	
Decay heat	ANS/ ANSI-5.1-1979	
Fuel rod peak linear heat generation ratio	44.0 kW/m	
Safety/Relief valve	Relief valve function	

Table 26.3-8 Initial Plant Conditions in Shutdown Modes

Items	Conditions	Remarks
Fuel operation cycle condition	17 month operation and 30 days outage	
Decay heat	ANS/ ANSI-5.1-1979	
Initial water level	RPV Normal Water Level to SFP Normal Water Level (Depends on the Operating State)	

Table 26.3-9 Initial Plant Conditions in SFP

Items	Conditions	Remarks
Decay Heat	ANS/ ANSI-5.1-1979 300 percent: Maximum Heat Load	
Initial water level	SFP Normal Water Level	
Initial temperature of SFP water	Maximum Heat Load condition	

26.3.5.3 Analysis Results

26.3.5.3.1 Beyond Design Basis Fault at Power

The analysis of the BDB faults at Power identified in Section 26.3.1.1 is reported in the topic report [Ref-41]. The results of all identified faults are shown in the topic report and the result of "Medium-term LOOP with CCF of EDGs and Failure of RCIC" is demonstrated as follows.

Medium-term (24 hours) LOOP is assumed as the initiating event, and CCF of EDGs and failure of Direct Current (DC) bus which supplies to RCIC is assumed as the additional failure. However, RDCF, FLSS, and Containment Venting are available as the countermeasures for this fault.

In the analysis result, reactor water level decreases due to loss of all ECCS and is lower than Top of Active Fuel (TAF) for a very short time, but it recovers after FLSS injection. PCV pressure increases due to SRV operation, but it decreases after Containment Venting. The plots of the key parameters for this event are shown in Figure 26.3-1 and Figure 26.3-2

The key results for this event are provided in Table 26.3-10. Peak cladding temperature is lower than 1,200 °C and cladding oxidation rate is lower than 15 percent. Reactor pressure is controlled by the SRV and is maintained below 120 percent of the maximum allowable working pressure. Due to the Containment Venting, it is found that the drywell and wetwell pressures can be maintained below the containment limiting pressure. The peak drywell temperature is also kept below the limiting value. Both peak pool temperature and peak wetwell temperature are also kept below the limiting value.

For dose evaluation, the bounding case of no fuel damage accident with scram where the maximum radionuclide is released is evaluated. The dose results are shown in Table 26.3-11. The dose results show that SP14.3.3 of the NSEDPs is met with considerable margin.

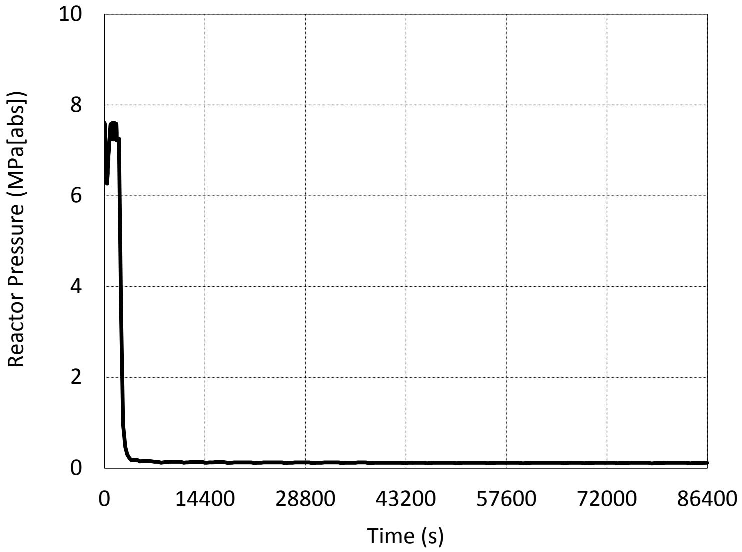
As shown the above, all of the acceptance criteria are satisfied.

Table 26.3-10 Key Outputs for Medium-term LOOP with CCF of EDGs and Failure of RCIC

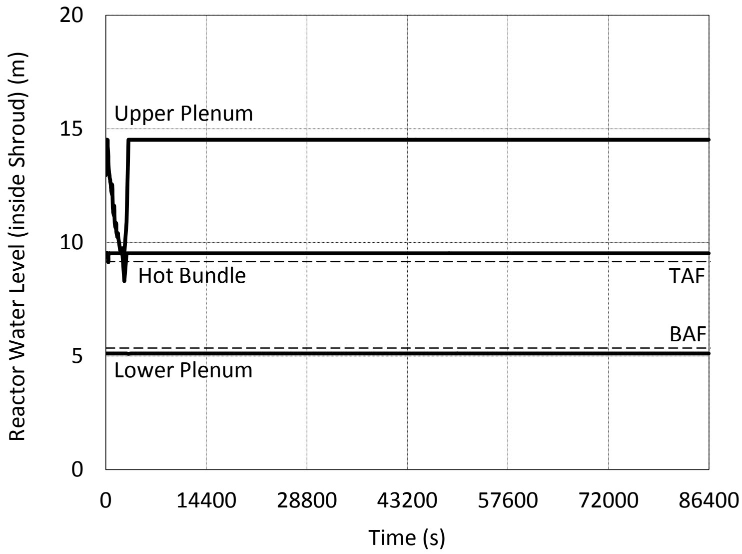
Parameters	Values
Peak Cladding Temperature	Approx. 320 (ºC)
Cladding Oxidation Rate	Approx. 0.1 (%)
Peak Reactor Pressure	Approx. 7.61 (MPa[abs])
Peak Drywell Pressure	Approx. 307 (kPa[gauge])
Peak Wetwell Pressure	310 (kPa[gauge])
Peak Drywell Temperature	Approx. 112 (ºC)
Peak Wetwell Airspace Temperature	Approx. 144 (ºC)
Peak Suppression Pool Temperature	Approx. 138 (ºC)

Table 26.3-11 Result of Dose (mSv)

1y age	10y age	Adult	Dose Criteria (SP14.3.3 of the NSEDPs)
2.6E+01	1.6E+01	1.8E+01	< 1,000

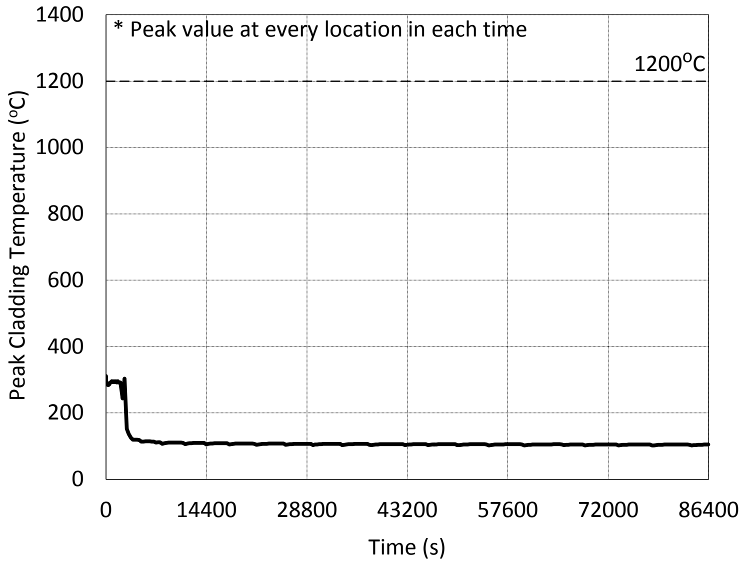


Reactor Pressure



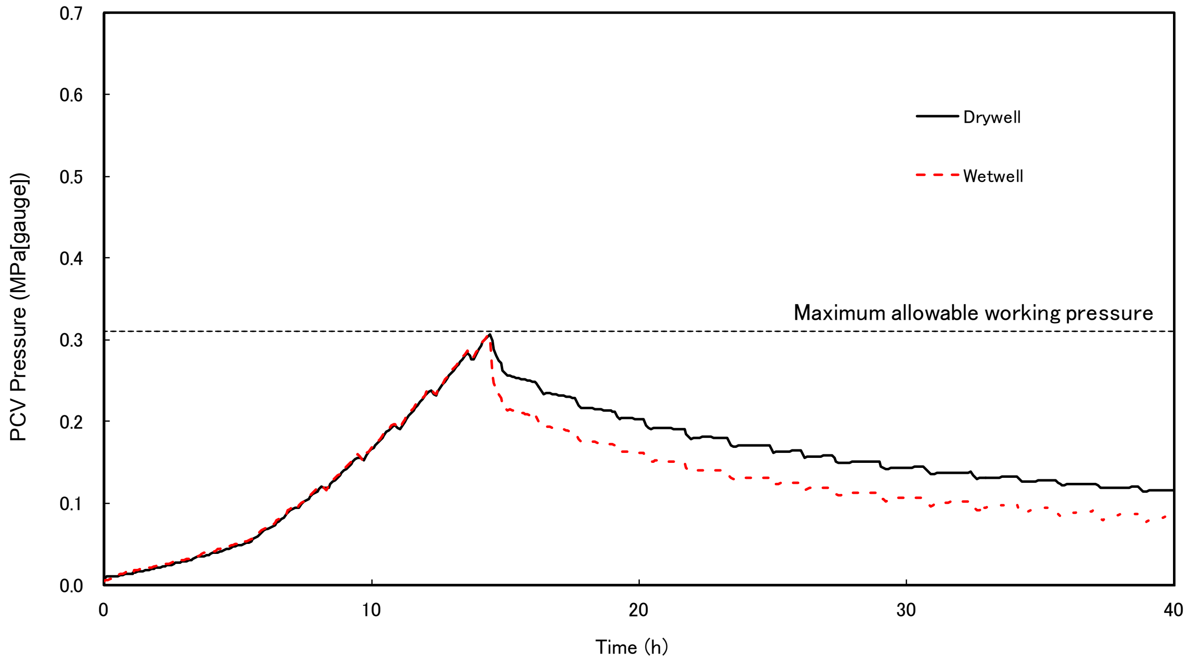
Reactor Water Level (inside Shroud)

Figure 26.3-1 Analysis Result of Reactor for Medium-term LOOP with CCF of EDGs and Failure of RCIC

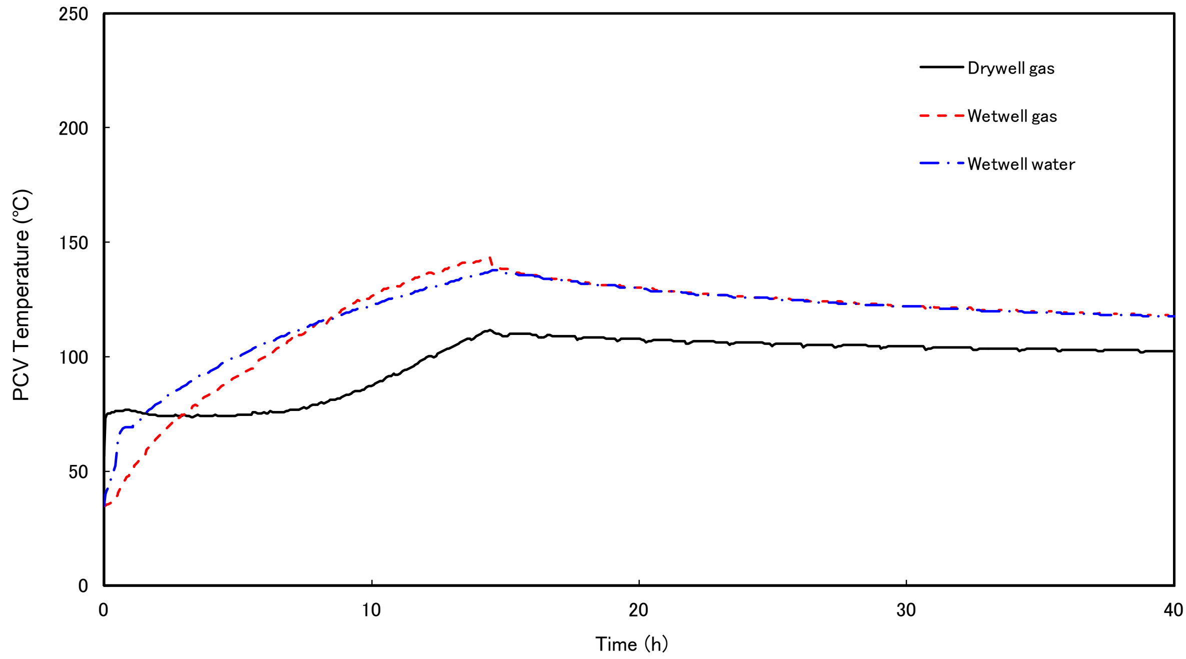


Peak Cladding Temperature

Figure 26.3-1 Analysis Result of Reactor for Medium-term LOOP with CCF of EDGs and Failure of RCIC (Cont'd)



(Containment pressure)



(Containment temperature)

Figure 26.3-2 Analysis Result of Containment for Medium-term LOOP with CCF of EDGs and Failure of RCIC

26.3.5.3.2 Beyond Design Basis Fault in Shutdown Mode

All of the BDB faults in shutdown modes identified in Section 26.3.2.2 are analysed. Here, in reality, the pressure boundary is at low pressure during shutdown modes and therefore the likelihood of LOCA events due to line break is extremely low. Therefore, the result of "Loss of operating RHR with Loss of all ECCS and Failure of FLSS (and FLSR)" is demonstrated as a representative fault sequence in this section. Moreover, the duration of each operating states is very short comparing to the duration of the entire plant operation. Therefore, "operating state C-3" (see Chapter 5, Section 5.4.3), in which duration is the longest in shutdown modes is demonstrated as the representative operating state in this section.

Loss of operating RHR in operating state C-3-1 is assumed as the initiating event, and Loss of all ECCS and failure of FLSS is assumed as additional failures. However, FLSR is available as defence in depth measures for this fault.

In the analysis result, the loss of RHR occurs and the operator is assumed to notice the occurrence of the loss operating RHR immediately. At 30 minutes after the occurrence of the initiating event, the operator tries to initiate standby RHR and FPC but they are assumed to fail. Then, the operator confirms availability of ECCS and FLSS but the operator can notice that all ECCSs and all FLSSs are not available for any reason in the next 30 minutes. Therefore, the operator starts preparation of FLSR at 60 minutes after the event. Line-up duration for FLSR is assumed to be 8 hours, and then the FLSR standbystate is achieved at 9.0 hours.. On the other hand, reactor well water temperature increases due to the loss of RHR, and then coolant boil-off occurs when reactor water temperature reaches 100 °C at 17.7 hours. Then, the operator starts the water injection by FLSR. Thereafter, the operator can maintain the water level at full of reactor well by the FLSR. This event sequence is shown in Table 26.3-12. In this case, reactor well water level does not decrease, and the reactor core and spent fuel remain covered during the event.

For dose evaluation, assumptions and methods used in BDB analysis in shutdown modes are the same as that for DB analysis in shutdown modes shown in Chapter 24, Section 24.10.2.1. Since the operator notifies the abnormity to the workers in the operating deck using paging before coolant boil-off starts, the workers can evacuate from the operating deck without any dose uptake. The dose results are shown in Table 26.3-13, and it shows that SP14.3.3 of the NSEDPs is met with considerable margin.

As shown the above, all of the acceptance criteria are satisfied.

Table 26.3-12 Event Sequence of Loss of Operating RHR with Loss of all ECCS and Failure of FLSS (Operating state C-3)

Time	Event
0 hour	· Loss of operating RHR occurs. · The operator notices occurrence of the event.
0.5 hours	· The operator tries to initiate standby RHR and FPC, but they fail. · The operator confirms ECCS and FLSS, but the operator notices their unavailability.
1.0 hours	· The operator starts preparation of FLSR.
9.0 hours	· Line-up of FLSR is completed before water temperature reaches 100 °C.
17.7 hours	· Water temperature reaches to 100°C and coolant boil-off starts. · FLSR injection is manually actuated and the reactor well water level is maintained. · The water level does not decrease from initial value.

Table 26.3-13 Result of Dose

Location	1y age	10y age	Adult	Dose Criteria (SP14.3.3 of the NSEDPs)
Off-site	5.9E-01 mSv	5.1E-01 mSv	6.0E-01 mSv	< 1,000 mSv
On-site (Control Room)	-	-	6.6E-02 mSv	< 2,000 mSv

26.3.5.3.3 Beyond Design Basis Fault for SFP and Fuel Route

All of the BDB faults in shutdown modes in identified Section 26.3.2.3 are analysed. In this section, the result of "Small Leak of SFP with Failure of FPC and FLSS" is demonstrated as the most severe case in terms of decrease of water level in SFP as follows.

Small Leak of SFP (liner leak) is assumed as the initiating event, and Failure of FPC and all FLSS is assumed as additional failures. However, FLSR is available as defence in depth measures for this fault.

Table 26.3-14 shows the event sequence. It is assumed that the operator notices the occurrence of leak and loss FPC when the SFP low water level alarm is triggered at 3.2 hours after the occurrence of initiating event. The operator tries to initiate standby FLSS at 30 minutes after the alarm, however, it is assumed that the operator notices that all FLSSs are not available for any reason in the next 30 minutes. Thus, the operator is assumed to start to prepare for FLSR operation 1 hour after the alarm. The completion time of FLSR preparation is assumed to be 8 hours and therefore the water injection by FLSR start at 12.2 hours. In this case, the lowest water level is evaluated to be about 6.0 m from TAF in the SFP and the spent fuel remains covered during the event.

As shown the above, all of the acceptance criteria are satisfied.

Table 26.3-14 Event Sequence of Small Leak of SFP with Failure of FPC and FLSS

Time	Event
0 hour	· SFP water level starts to decreases due to small leak. · Loss of operating FPC occurs.
3.2 hours	· The operator notices the occurrence of the event.
4.2 hours	· The operator starts preparation of FLSR.
9.1 hours	· Water temperature reaches 100 °C and water evaporation starts.
12.2 hours	· FLSR injection is manually actuated. · Lowest water level is about 6.0 m from TAF in SFP.
14.0 hours	· Water level is recovered to the normal operating water level.

26.4 Severe Accident Analysis

Severe accidents are defined as those fault sequences that could lead either to consequences exceeding the highest off-site radiological doses given in the BSL of the NSEDPs (SP14.2.1), or to an unintended relocation of a substantial quantity of radioactive material within the facility which places a demand on the integrity of the remaining physical barriers [Ref-12] [Ref-17]. Severe accident analysis has been performed to analyse those fault sequences that are likely to give rise to these consequences. Severe accident analysis is performed in a best-estimate manner. The objectives of these severe accident analyses are;

- To understand the severe accident phenomena and progression, including fission product release and transport behaviour.
- To determine the time margin for accident management such as equipment recovery and operator action.
- To evaluate engineered features, strategies, and procedures for mitigating a severe accident.
- To demonstrate the effectiveness of engineered features, strategies, and procedures.
- To determine the magnitude and characteristics of the predicted source term as the input of potential radiological consequences, including societal risk.
- To demonstrate that there is no sudden escalation of consequences just beyond the design basis.
- To support Level 2 PSA quantification.

The section provides a discussion on the application of severe accident analysis with respect to the aims listed above.

26.4.1 Severe Accident Safety Case Strategy

(1) Approach on Severe Accident Analysis Application

This section identifies the application areas and how the severe accident analyses are used to evaluate each application area. Severe accident analysis application areas are shown below and are summarised in Figure 26.4-1**;**

- ALARP demonstration of preventive/mitigating measures,
- Technical basis for accident management strategy,
- Development of emergency plans,
- Source terms for Level 3 PSA and radiological consequences assessment,
- Effectiveness of countermeasures,
- Understanding of the severe accident progression and phenomena, and

- Demonstration on Cliff Edge Effect.

(a) ALARP demonstration of Preventive/Mitigating Measures

The severe accident analysis considers high level strategy, engineered severe accident features and the relevant procedures. By demonstrating that the plant is brought to and maintained in a stable condition, the effectiveness of countermeasures is confirmed for specific PDSs (Plant Damage States) (PCSR Chapter 25 (Probabilistic Safety Assessment)) in combination with countermeasures through mitigated sequence analysis. ALARP justification for severe accident countermeasures is shown in Section 26.7.

(b) Technical Basis for Accident Management Strategy

Mitigated sequence analysis considers the accident management strategy, which is described in Section 26.4.3, by using the MAAP code. Energetic phenomena such as steam explosion, which are not modelled in MAAP, are evaluated separately (see Section 26.4.4). These analyses demonstrate that the accident management strategy for the UK ABWR has the basis to control a severe accident condition.

(c) Development of Emergency Plans

The Severe Accident Analysis provides the source terms, which are used by defining the input for Level 3 PSA and the off-site radiological assessment, as outlined in (d). Preparation of emergency plans for the protection of people is considered based on the radiological consequences of Level 3 PSA. Magnitude and characteristics of radiological consequence is obtained using the MAAP code, which is described in Section 26.4.4.1. Emergency preparedness is described in PCSR Chapter 22 (Emergency Preparedness).

(d) Source terms for Level 3 PSA and Radiological Consequences assessment

Source term analysis provides the magnitude and duration of fission product release, which are then used as the input for Level 3 PSA. The Level 3 PSA analyses the radiological consequences. Source term analyses for at-power operation, shutdown, and the SFP are shown in Sections 26.4.6.3, 26.4.7, and 26.4.8, respectively.

(e) Effectiveness of countermeasures

Severe accident analysis selects a representative plant condition and demonstrates the effectiveness of countermeasures with regard to the plant response (e.g. stable thermal hydraulic condition, fission product release reduction). Basic design of severe accident measures is described in Section 26.4.3.4. Effectiveness of these countermeasures is confirmed for specific PDSs (which are defined in PCSR Section 25.5.1 (Interface between Level 1 and Level 2 PSA)) in combination with countermeasures through mitigated sequence analysis. Severe accident analyses in mitigated sequences for at-power operation, shutdown, and the SFP are described in Section 26.4.6.2, Section 26.4.7, and Section 26.4.8, respectively.

(f) Understanding of the Severe Accident Progression and Phenomena

Appropriate understanding of severe accident phenomena and progression is important to prevent and/or mitigate a severe accident in the UK ABWR. Severe accident progression and severe accident phenomena that could hypothetically occur in the UK ABWR is described in Section 26.4.2. Severe accident analyses in unmitigated sequences for at-power operation, shutdown, and the SFP are described in Sections 26.4.6.1, 26.4.7, and 26.4.8, respectively. Effectiveness of these countermeasures is confirmed for specific PDSs in combination with countermeasures through mitigated sequence analysis. Some severe accident phenomena that could hypothetically occur as the severe accident progresses will lead to the containment being challenged. Therefore, PCV failure probability evaluation for three failure modes, Molten Core Concrete Interaction (MCCI), Fuel Coolant Interaction (FCI) and Direct Containment Heating (DCH), have been evaluated considering the uncertainties of the dominant parameters for each failure mode. It has been described in Section 26.4.5.

(g) Demonstration on Cliff Edge Effect

Effectiveness evaluation of engineered features, strategies and procedures are used for this demonstration. Severe accident analysis in a mitigated sequence for at-power operation, shutdown, and the SFP are described in Sections 26.4.6.2, 26.4.7, and 26.4.8, respectively.

(2) Selection Criteria for Postulated Initiating Events and Loss of Safety Functions

The analysis condition in a severe accident defines the initiating events and the subsequent loss of safety functions. The selection of initiating events and the loss of safety functions are based on the characteristics of the plant response and the effect on fission product releases. Key items to select these are identical to those in the PDSs identification process, which is described in PCSR Section 25.5.1 (Interface between Level 1 and Level 2 PSA). The selection of initiating events and the loss of safety functions are based on the characteristics of the plant response and the effect on fission product releases, which are as follows;

- Loss of feedwater flow is selected as an initiating event for some accident sequences. This leads to both an early core damage and RPV failure.
- Closure of MSIVs is selected as an initiating event for some accident sequences where early PCV failure leads to early core damage (e.g. TC). This leads to both an early core damage and RPV failure.
- Break/isolation failure of RHR line is selected as a representative location for some accident sequences where LOCA leads to immediate RPV depressurisation. As the RHR line has a

large cross-sectional area and is installed in the low elevation among piping in the RPV, this leads to both an early core damage and RPV failure.

- Break/isolation failure of bottom drain line is selected as a representative location for some accident sequences where LOCA does not lead to immediate RPV depressurisation. As the bottom drain line is the largest pipe where immediate depressurisation does not occur, this leads to both an early core damage and RPV failure.

(3) Criteria for Effectiveness Evaluation

Effectiveness evaluation demonstrates that the plant is brought to and maintained in stable condition after a severe accident. Table 26.4-1 shows severe accident phenomena and the associated success criteria conditions. Criteria of success for overpressure and overtemperature of the containment are shown below;

- The plant does not experience the PCV failure condition as shown in Figure 26.4-4,
- Core or corium is covered by water or steam,
- Containment pressure is decreasing or maintained under almost saturated condition,
- Containment temperature is decreasing or maintained under almost saturated condition, and
- It is judged that above conditions are maintained for a period of over 24 hours.

The basis of PCV failure condition due to overpressure and over temperature is described in PCSR Section 10.6.4 (Severe Accident Analysis) and Section 25.5.2.3 (Containment Performance Analysis).

Success criteria for other containment failure modes, such as Direct Containment Heating (DCH), hydrogen combustion, ex-vessel Fuel Coolant Interaction (FCI), Molten Core Concrete Interaction (MCCI) and direct contact of corium (containment shell attack), are shown below;

- DCH:
  RPV is depressurised to lower than 2.0 MPa [gauge] [Ref-14] before RPV failure occurs.
- Hydrogen combustion:
  Hydrogen concentration is controlled under 4 percent or the oxygen concentration is controlled under 5 percent when steam concentration is under 60 percent. When steam concentration is over 60 percent, hydrogen combustion does not occur [Ref-15].
- Ex-vessel FCI:
  Supporting function of the reactor vessel is not lost due to the impulse load to the pedestal wall by ex-vessel FCI nor does the PCV boundary fail due to rapid pressurisation as a result of ex-vessel FCI. Specifically, the maximum yield stress produced in the steel plate of the pedestal wall due to the dynamic load of the ex-vessel FCI (steam explosion) is less than the yield stress of the steel plate. Rapid pressurisation as a result of ex-vessel FCI is less than 620 kPa [gauge], which is the same as the criterion for PCV overpressure.

- MCCI:
  Supporting function of the reactor vessel is not lost due to the erosion of pedestal wall by MCCI. Specifically, the erosion depth does not reach the ultimate erosion depth where the supporting function of the outer steel plate is lost due to the increasing temperature.
- Direct contact of corium:
  PCV boundary does not fail as a result of direct contact with corium (containment shell attack). Direct contact of corium (containment shell attack) in RPV failure scenario with low pressure does not need to be considered in the UK ABWR. As the lower drywell is surrounded by the pedestal wall, corium does not come into direct contact with the PCV boundary (i.e. liner, hatch, and etc.) in the UK ABWR.

(4) Emergency Procedures

Outline of emergency procedures is described in Section 26.4.3. The severe accident analyses are conducted based on the emergency procedures for mitigated sequences, which are described in Sections 26.4.6.2, 26.4.7, and 26.4.8, respectively.

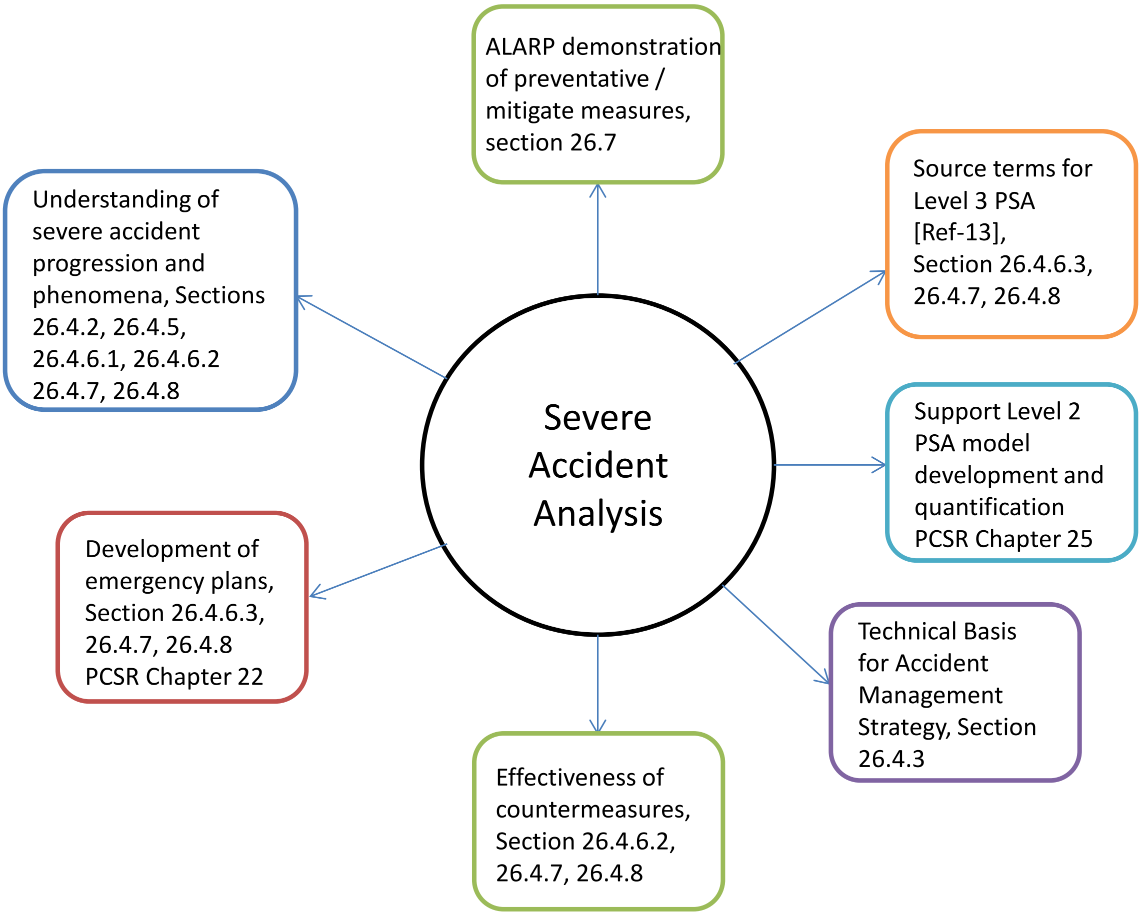


Figure 26.4-1 Application of Severe Accident Analysis

Table 26.4-1 Phenomena and Conditions of Success Criteria

Phenomena	Conditions of success criteria
Overpressure and overtemperature of the containment	- The plant does not experience the PCV failure condition as shown in Figure 26.4-4. - Core or corium is covered by water or steam. - Containment pressure is decreasing or maintained under almost saturated condition. - Containment temperature is decreasing or maintained under almost saturated condition. - It is judged that above condition is maintained for over 24 hours.
Direct Containment Heating (DCH)	RPV is depressurised to lower than 2.0 MPa [gauge] [Ref-14] before RPV failure occurs.
Hydrogen combustion	- When steam concentration is under 60 percent, hydrogen concentration is controlled under 4 percent or the oxygen concentration is controlled under 5 percent. - When steam concentration is over 60 percent, hydrogen combustion does not occur. [Ref-15]
Ex-vessel Fuel Coolant Interaction (FCI)	- The maximum yield stress produced in the steel plate of the pedestal wall due to the dynamic load of the ex-vessel steam explosion (FCI) is less than the yield stress of the steel plate. - Rapid pressurisation as a result of ex-vessel FCI is less than 620 kPa [gauge], which is the same as the criterion for PCV overpressure.
Molten Core Concrete Interaction (MCCI)	The success criterion for the prevention of PCV failure due to MCCI is that the erosion depth of pedestal wall does not reach the ultimate erosion depth where the supporting function of the outer steel plate is lost due to the increasing temperature.

26.4.2 Severe Accident Progression and Phenomena

26.4.2.1 Severe Accident Progression

Severe accident progression is categorised into three phases. The first phase is called the in-core phase, which is defined as the period from the accident initiation to core support plate failure. The second phase is called the lower plenum phase, which is defined as the period from core support plate failure to RPV failure. The third phase is referred to as the ex-vessel phase, which is defined as the period after RPV failure. An overview of the three phases is provided in Figure 25.5-2 of Chapter 25. The following is the outline of severe accident progression for each phase.

(1) In-core phase

If a transient event occurs during at-power operation, the control rods are automatically inserted to the core and the PCV boundary is isolated automatically. Although there is a risk of the scram failure, reactivity in the core can be controlled by the ARI (Alternate Rod Insertion) and the Standby Liquid Control system (SLC). If the reactivity control is successful, the reactor becomes sub-critical and reactor power promptly reduces to a few percent of the rated power. Then, cold shutdown state can be achieved by using the RHR/LPFL. Since the RHR/LPFL has three independent and redundant divisions, the core cooling can be attained with high reliability. However, if the RHR/LPFL fails and other emergency core cooling systems such as HPCF and RCIC fail, the reactor water level gradually decreases and it might lead to a severe accident.

Even if reactor water level decreases due to the failure of RHRs and other ECCS, alternative water injection systems such as the FLSS can be used to prevent the core damage. However, if these systems fail, reactor water level decreases below the Top of Active Fuel (TAF) and fuels are exposed from the water surface.

In such as situation, the fuel and cladding temperature above the water level increase, and hydrogen is generated due to the metal-water reaction. Then, fuel damage occurs due to high temperature of the cladding and fission products are released from the damaged core. If alternative water injection from FLSS or FLSR is available, the core can be re-flooded and cooled. When the damaged core is re-flooded, additional hydrogen is generated due to metal-water reaction. As control rods may melt before the fuel, re-criticality may occur during the re-flooding phase, but this risk is judged to be small as shown in Section 26.4.2.2. If alternative water injection is failed, damaged fuel melts and relocates to the lower regions of the core. Mixtures of molten fuel and core structural materials are called corium, and solidified corium is called crust. If the core support plate fails due to the heat of the corium, it will relocate into the lower plenum.

(2) Lower plenum phase

When the core support plate is damaged, the corium relocates into the lower plenum through the failure area of the core support plate. As there is water pool in the lower plenum, corium particles are generated due to break-up effect when it falls into the lower plenum. In addition, there is the risk of energetic fuel coolant interaction in the lower plenum pool, which is called In-Vessel FCI. However, the risk of In-Vessel FCI (steam explosion) is judged to be negligibly small because the energy generated by FCI is limited due to energy absorption by the internal pressure vessel structures [Ref-2].

The corium accumulating in the lower plenum forms a layer of particulate corium (particulate debris bed), a metal layer with the low density and molten corium which is surrounded by the solidified corium (crust). Heat transfer between the RPV lower head and the adjacent crust is restricted by a gap between the crust and the RPV lower head. The gap expands by creep deformation of the RPV lower head. As a result, water in the lower plenum flows into the gap and corium cooling is maintained (gap cooling phenomena). After the water in the lower plenum is depleted, the corium heat is transferred by the radiation, and the temperature of the structural materials in the lower plenum increases. Then, the structures in the lower plenum, such as the RPV lower head, CRD guide tube, CRD housing tube, Instrumentation tube and housing, heats up, and finally RPV failure occurs. From the result of the MAAP calculation, the ejection of CRD tubes are identified as a dominant RPV failure mode in the UK ABWR.

After the RPV lower head fails, the corium is discharged into the lower drywell through the failure opening. The failure opening gradually expands as a result of the high temperature outflow of the corium, which is called ablation effect.

(3) Ex-vessel phase

In most of scenario, RPV failure occurs with low pressure because accident management guide directs operators to depressurise the RPV and reliable depressurisation systems such as SRVs and RDCF are installed in the UK ABWR. However, in some scenario, RPV failure with high pressure might occur and lead to High-Pressure Melt Ejection (HPME). If HPME occurs, the corium jet is disrupted by high velocity gas flow and fragmented corium droplets are generated. Then, the fragmented corium droplet might move from the lower D/W to the upper D/W. As the corium droplets have a large surface area per volume and heat transfer from the corium to the gas is large, the PCV is pressurised rapidly and it might lead to PCV failure. These phenomena are called DCH.
In the UK ABWR, water injection to the lower D/W before RPV failure is planned to be conducted to enhance the corium cooling in the lower D/W and mitigate the MCCI. If pre-injection is

conducted and water pool is formed in the lower drywell, the corium jet breaks up in the lower D/W pool and the coolability of the corium is significantly enhanced because a particulate debris bed is generated. On the other hand, pre-water injection might lead to ex-vessel FCI (steam explosion). However, as the pedestal wall is made of thick steel structures and concrete, it is expected that the pedestal failure is unlikely to occur even if ex-vessel FCI (steam explosion) occurs in the lower D/W. Even if pre-water injection into the lower D/W fails, the corium in the lower D/W can be cooled passively by the activation of the Lower Drywell Flooder system (LDF). The LDF has a fusible plug and piping which connects the lower D/W and the suppression pool. When the gas temperature in the lower D/W increases due to the heat of the corium, the fusible plug melts passively and water is supplied from the suppression pool to the lower D/W. Then, the corium in the lower D/W is cooled. After the initial quench of the corium, water in the lower D/W gradually evaporates from the decay heat of the corium. The steam flows into the suppression pool. Although the steam is condensed in the suppression pool, the pool gradually heats up and PCV pressure increases.

To prevent the PCV failure due to overpressure, two kinds of PCV venting systems are installed in the UK ABWR. One is the Filtered Containment Venting System (FCVS) and the other is the Hardened Containment Venting System. Generally, containment venting is conducted using the FCVS via the W/W because the most of fission products are expected to be scrubbed into the suppression pool and the scrubber tank of the FCVS.

Even if the manual PCV venting is failed, the containment venting is conducted passively because a rupture disk is installed in the FCVS. This is called the Containment Overpressure Protection System (COPS) and the rupture disk is passively opened before the PCV pressure reaches the ultimate pressure of 620 kPa[gauge].

26.4.2.2 Severe Accident Phenomena

The followings are identified as important phenomena to be considered in a severe accident:

(1) Overpressure and overtemperature

In severe accidents, PCV temperature increases due to the heat generation from the chemical reaction of the metal-water reaction and decay heat from fission products. Simultaneously, PCV pressure increases due to the non-condensable gas generation by MCCI, hydrogen generation in the metal-water reaction, and steam generation due to decay heat from fission products. As the result, PCV pressure and temperature reaches the ultimate capacity of the containment and then PCV failure might occur due to overpressure and overtemperature.

In the UK ABWR, various kinds of water injection systems such as ECCS, FLSS, and FLSR are installed and various kinds of water injection lines (Core, lower drywell, upper D/W spray, and reactor well) are prepared to mitigate PCV temperature increasing. If these systems are available, PCV temperature can be controlled under the ultimate temperature. In addition, Containment Overpressure Protection System (COPS), which is the system for passive containment venting, is installed to prevent PCV failure due to overpressure. Therefore, it is judged that the risk of PCV failure due to overpressure and overtemperature is decreased to as low as reasonably practicable in the UK ABWR.

Effectiveness evaluations for core injection, PCV spray, and PCV venting are summarised in Section 26.4.6.2.

(2) Molten Core Concrete Interaction (MCCI)

After RPV failure occurs, corium is discharged from the lower plenum into the lower D/W. If corium cooling is insufficient, the corium thermally decomposes and erodes the concrete, and then water (H_2O) and carbon dioxide (CO_2) are released from the concrete. When the H_2O and CO_2 rise through the corium pool, they cause chemical reactions with the high temperature corium and are deoxidized to hydrogen (H_2) and carbon monoxide (CO), which are combustible gases. In addition, if MCCI continues for a long time, pedestal failure occur and the supporting function of the RPV might be lost. As the result, PCV failure might occur because RPV is shifted lower and the piping connecting to the PCV boundary might fail.

As the countermeasure of the MCCI, the UK ABWR has following design and systems.

- Large spreading area in the lower D/W
  The UK ABWR is designed to ensure there is sufficient spreading area for the corium cooling in the lower D/W. The lower D/W has large flat floor and the diameter of the lower D/W is more than 10 m. Even if pre-injection to the lower D/W fails, the corium spreads throughout the lower drywell floor and it enhances the heat transfer from the surface of the corium when water is injected to the lower D/W.
- Water Injection System for Corium Cooling in the lower D/W
  In the UK ABWR, pre-injection to the lower D/W before RPV failure is conducted to enhance the corium coolability in the lower D/W. The FLSS and the FLSR can be used for this operation. If pre-injection is conducted and water pool is formed in the lower drywell, the corium jet breaks up in the lower D/W pool and the coolability of the corium is enhanced because a particulate debris bed is generated. Even if pre-water injection into the lower D/W fails, the corium in the lower D/W can be cooled passively by the activation of the Lower Drywell Flooder system (LDF).

- The pedestal wall and sacrificial layer of the lower D/W floor
  To protect the pedestal wall and the liner in the lower D/W floor, thick concrete exists in the pedestal wall and above the liner of the lower D/W floor. The thickness of concrete is approximately 1.6 m. The concrete is made with low gas content material to minimise pressurisation due to the gases generated in a MCCI event.
- Corium Shield
  Corium shield is installed to prevent deep pools of corium from forming in the sump region. The two sumps in lower D/W are needed for collecting drain water and detecting leakage flow during normal operation. The corium shield has channels to allow water in the lower D/W to flow into the sump. The length of channels is long enough for any molten debris to freeze before it reaches the exit of channels and spills into the sump. Corium shield is made of ceramic refractory material, which has high melting point and chemical stability.

From the above considerations, it is judged that the risk of PCV failure due to MCCI is decreased to as low as reasonably practicable in the UK ABWR. However, to consider the small risk of the PCV failure due to MCCI, PCV failure probability due to MCCI is evaluated as shown in Section 26.4.5 and it is considered in the Level 2 PSA.

(3) Fuel Coolant Interaction (FCI)

Challenges to the containment during a severe accident may result from Fuel Coolant Interaction (FCI). Fuel coolant interactions are most likely to challenge the containment when molten debris falls into water. Both the impulse and static loads are considered. Fuel coolant interactions may occur either at the timing of core support plate failure when corium falls from the core to the lower plenum or at the timing of RPV failure when corium falls from the lower plenum to the lower D/W. With regard to in-vessel FCI (α-mode PCV damage), most experts acknowledge that the risk of in-vessel FCI is extremely low [Ref-2]. However, it is considered in Level 2 and Level 3 PSA as one of release categories and the source term analysis is conducted for this scenario.

With regard to ex-vessel FCI, steam explosion due to the contact of water and corium is also unlikely to occur. For ex-vessel FCI, many large-scale experiments and studies have been conducted since the 1970's. In the FARO test, the KROTOS test and the COTELS test, which are experiments using UO_2 corium, steam explosion did not occur except for some cases where shock waves is given to the corium from outside as an external trigger [Ref-39] [Ref-3]. In addition, even if steam explosion occurs in the lower drywell, the failure probability of the pedestal wall is very small because the pedestal wall is made of thick concrete and steel structures. Therefore, it is judged that the risk of PCV failure due to ex-vessel and in-vessel FCI is decreased to as low as reasonably practicable in the UK ABWR. However, to consider the small risk of the PCV failure due to ex-vessel FCI, PCV failure probability due to ex-vessel FCI is evaluated as shown in Section 26.4.5 and it is considered in the Level 2 PSA.

(4) Direct Containment Heating (DCH)

If RPV failure occurs with high pressure, the corium may be entrained out of the lower drywell. If the corium is finely fragmented, the pressure and the temperature in the containment increase rapidly. This process is called Direct Containment Heating (DCH). The magnitude of the pressure increase is dependent on the amount of corium involved in the event. If a large fraction of corium participates are generated in the DCH event, the containment may be challenged. The probability of the DCH occurring is extremely small for the UK ABWR for two reasons. First, the UK ABWR has many diverse measures to depressurise the RPV such as SRVs and RDCF. The second reason is the UK ABWR has ten connecting venting pipes which have small flow area. Although they are connected from the lower drywell to upper drywell and wetwell, large part of corium particles and steam flow into wetwell when RPV failure occurs with high pressure. Therefore, it is judged that the risk of PCV failure due to DCH is decreased to as low as reasonably practicable in the UK ABWR. However, to consider the small risk of the DCH, PCV failure probability due to DCH is evaluated as shown in Section 26.4.5 and it is considered in the Level 2 PSA (Section 25.5.2.1).

(5) Hydrogen Combustion

Hydrogen is generated by the metal-water reaction, Molten Core Concrete Interaction (MCCI) and radiolysis of water. If oxygen exists in the PCV, hydrogen could burn and release heat. This could pressurise the PCV and potentially leads to the PCV failure.

In the UK ABWR, the containment is inerted with nitrogen most of time during at-power operation. Therefore, even if severe accident occurs during at-power operation and hydrogen is generated due to metal-water reaction in the PCV, hydrogen combustion does not occur in the PCV because oxygen concentration is low in the PCV. In addition, as Passive Autocatalytic Recombiners (PARs) are installed in the PCV, oxygen and hydrogen gases generated from the radiolysis of water are passively recombined by the PARs.

In the case of an accident when the containment is not inerted or the reactor is shutdown or in the SFP, hydrogen management is important because hydrogen gas is released directly to the refueling floor which is not inerted with nitrogen. Generally, as the large volume of water present above the fuel in the reactor well during shutdown or in the SFP, there is a large time margin available to prevent fuel damage. Therefore, most of accident in the shutdown reactor and in the SFP can be terminated without fuel damage and no hydrogen is generated. In addition, even if severe accident occurs in the shutdown reactor and the SFP (Note: it is thought that these accidents are practically eliminated), hydrogen in the refueling floor can be removed by using PARs, Standby Gas Treatment System (SGTS), and blowout panel.

From the above considerations, it is judged that the risk of hydrogen combustion is decreased to as low as reasonably practicable in the UK ABWR (See Section 26.7.7).

(6) Re-criticality during In-vessel Recovery

If water injection is conducted after the control blades melt, there is small possibility for re-criticality to occur within small time window during an in-vessel recovery. This is due to the potential for boron carbide in the control blades to form a eutectic with steel at 1500 K and relocate, whereas the fuel starts to melt 2500 K. As a consequence, there might be regions of the core without control rods which may result in core re-criticality when water is injected in a short time after the core damage. This could increase the steam generation rate and potentially lead to the PCV failure due to overpressure. However, the re-criticality during in-vessel recovery is unlikely to occur from the following reasons.

- If core injection from FLSS is conducted within the first 30 minutes after the start of an accident, re-criticality does not occur because a large degree of core damage and the melting of control rods would not have occurred.
- If core injection from FLSS is recovered within the time window of 30 to 40 minutes after the start of an accident, there is a possibility that the control rods may have melted and relocated but the fuel remains in the core. However, the probability of recovering core cooling in this short interval is significantly small. In order for re-criticality to occur, there must be system or operator failure, then water must be injected in the time window of about 10 minutes. Based on the standard models for recovery of systems and operator error, it is concluded that the probability of this occurrence is small. In addition, the emergency procedures direct the operator to inject water with boron from the SLC system (if possible). If boron injection is conducted, the re-criticality event does not occur. SLC will be available in most of this scenario because SLC is also supported by B/B class 2 electrical power supply system, which supports FLSS.

From the above considerations, it is judged that the risk of PCV failure due to re-criticality during in-vessel recovery is decreased to as low as reasonably practicable in the UK ABWR.

26.4.3 Severe Accident Management Strategy

26.4.3.1 Severe Accident Management Strategy

Severe Accident (SA) measures are designed to prevent or mitigate fuel damage in a reactor accident or in a spent fuel pool accident. They are also designed to prevent/mitigate containment failure and a large amount of fission products being released to the environment during severe accident. The following is the severe accident management strategy in the UK ABWR:

(1) Core cooling strategy

If all of core cooling measures and core injection measures are lost, severe accident progresses as described in Section 26.4.2.1. In the severe accident management strategy for the UK ABWR, core injection using alternative water injection systems (i.e. FLSS or FLSR) is prioritised at the initial stage of the accident to terminate severe accident progression. In the very unlikely event that the RPV depressurisation function fails and RPV pressure is still high, alternative depressurisation system (i.e. RDCF) is actuated and core injection by using alternative low pressure injection system is accomplished. By using these countermeasures, core damage is prevented/mitigated in most accident scenarios including if the RPV water level decreases rapidly, for example, due to a large break LOCA. If core injection is conducted, the core is cooled and the possibility of corium relocation to the lower plenum is reduced. On the basis of the strategy described above, procedures are prepared so that operator can appropriately respond to an accident based on information obtained from instrumentation that can measure, monitor, and record the plant data in severe accident.

(2) Containment control strategy

Containment control strategy has two parts. The first part is the flooding strategy which is intended to maintain containment boundary integrity. The second part is the heat removal strategy which is intended to remove the decay heat from the containment. Procedures for these strategies are prepared so that operator can appropriately respond to accident based on information obtained from instrumentations which measure, monitor, and record the plant data in a severe accident.

(a) Flooding strategy for containment

In the case where core cooling is successful to terminate the severe accident progression, the containment is pressurised due to steam and hydrogen generated in the core. At the same time, PCV temperature increases due to the decay heat from fission products released into the PCV. Even if PCV spray from RHR/LPFL system fails, alternative upper D/W spray from FLSS or FLSR is available to control the PCV temperature and pressure and decrease the risk of overtemperature and overpressure failure of the PCV.

In the case where core cooling fails or is delayed, the core is relocated to the lower plenum and the melted core heats the RPV lower head structures. In the severe accident management for the UK ABWR, lower D/W injection is conducted before RPV failure occurs and water pool is formed in the lower D/W to enhance the corium coolability and mitigate MCCI. Furthermore, RPV is depressurised before RPV failure to prevent the occurrence of High-Pressure Melt Ejection (HPME), as described in Section 26.4.2.2.

In the case where core cooling fails and RPV failure occurs, the corium in the lower plenum is discharged from the RPV and spreads onto the lower D/W floor. As the UK ABWR has a large floor area in the lower D/W, the corium can be effectively cooled and concrete erosion due to MCCI is mitigated. Alternative water injection systems from FLSS or FLSR can supply water into the lower D/W and maintain corium flooding. In the very unlikely event where water injection into lower D/W fails or is delayed, passive equipment which is called the Lower Drywell Flooder System (LDF) is actuated when the temperature of the lower D/W increases and it supplies water into the lower D/W. After the RHR/LPFL is restored it can maintain core and containment cooling without alternative water injection.

(b) Heat removal strategy for containment

Heat removal from the containment is important to prevent PCV failure due to overpressure. Heat removal from the containment is attained using RHR/LPFL system or containment venting system. The following is the basic policy for the containment heat removal:

- The first priority of containment heat removal operation is the use of RHR/LPFL system.
- The second priority of containment heat removal operation is the recovery of RHR/LPFL system.
- For extremely severe event such as LOOP plus multiple failures of safety systems, water injection to the reactor plus filtered containment venting through the suppression pool is considered as containment heat removal measures to prevent the containment failure due to overpressure.

When RHR/LPFL systems have failed in the event of a “LOOP + Class 1 EDGs Failure”, the RHR/LPFL system can be recovered using the Diverse Additional Generator (DAG) or large power trucks. When the RHR/LPFL systems have failed in the event of a “Loss of Ultimate Heat Sink (LUHS)”, the RHR/LPFL system can be recovered using a Reserve Ultimate Heat Sink (RUHS) or an Alternate Heat Exchange Facility (AHEF). The RUHS is deployable on timescales quicker than the AHEF.

If RHR/LPFL cannot be recovered using the above systems, filtered containment venting is conducted manually to prevent PCV failure due to overpressure. Even if containment venting by operator action fails, it is passively activated by a rupture disc installed in the Filtered Containment Venting System (FCVS) when the pressure inside the containment reaches the design setpoint pressure.

(3) Fission product release control strategy

As described in “Core cooling strategy” and “Containment control strategy”, the basic strategy to mitigate fission product release to the environment is to provide core and corium cooling as well as containment cooling in order to maintain containment boundary integrity. In the severe accident management strategy for the UK ABWR, the timing of PCV venting is delayed as much as possible to facilitate RHR recovery and to decrease public dose. Even if PCV venting is conducted, it is conducted through the W/W because the most of fission products are expected to be scrubbed into the suppression pool. In addition, the UK ABWR is equipped with the FCVS which can further reduce the amount of fission products released to the environment.

(4) Spent fuel pool cooling strategy

If the cooling function (FPC, RHR) and makeup function (MUWC, SPCU, Fire Pump) of the Spent Fuel Pool (SFP) fail, alternative water injection systems from FLSS or FLSR are used to inject water into the SFP, keeping the water level of the SFP above TAF and cooling the fuel in the SFP. In the very unlikely event the SFP water level cannot be maintained due to a large break of the SFP structure, SFP spray can prevent/mitigate severe spent fuel damage and suppress the amount of fission product released from SFP.

(5) Hydrogen control strategy

In the UK ABWR, the containment is filled with nitrogen most of time during at-power operation to ensure that oxygen concentration is controlled under the flammability limit even if flammable gases are generated due to the metal-water reaction, Molten Core Concrete Interaction (MCCI) and radiolysis of water in the containment. In addition, due to Passive Auto-catalytic Recombiner (PAR) being installed in the containment, oxygen and hydrogen gases generated from the radiolysis of water are passively recombined.

In the case of an accident when the containment is not inerted or the reactor is shutdown, hydrogen management is important. Generally, as large volume of water is present above the fuel in the reactor well and core power is low, there is large time margin available to prevent fuel damage. Therefore, prevention of hydrogen gas generation by water injection is the most important countermeasure in this period.

In a very unlikely event that severe accident occurs in the shutdown reactor, hydrogen in the refueling floor is controlled using the following systems:

- PCV boundary,
- Standby Gas Treatment System (SGTS),
- PARs in the reactor building, and
- Blowout panel and entrance door for large equipment.

More detailed ALARP discussion on flammable gas control is summarised in Section 26.7.7.

26.4.3.2 Basic Requirements for Severe Accident Management Measures

Severe accident (SA) management measures are based on the basic requirements below:

- SA management measures are designed to function as required with high reliability under a severe accident condition.
- SA management measures are designed to be controlled with high reliability under a severe accident condition.
- SA management measures are designed to be capable of inspection and testing.
- SA management measures are designed so as not to cause any detrimental impact on other equipment.
- SA management measures are designed to have sufficient capacity for severe accidents.
- SA management measures are designed so that diversity is considered as much as possible for the Design Basis Accident management measures.
- SA management measures are designed to have a resistance against external hazards.
- Mobile SA management measures are designed to provide reliable connection to permanent equipment, and multiple connections are prepared with appropriate spatial separation to avoid disconnection due to common failure modes.
- SA management measures involving manual operator actions are designed with human factors engineering consideration in order to achieve a high reliability.

Among the above requirements, "sufficient capacity for severe accidents" is confirmed by the effectiveness evaluation described in Section 26.4.6.2. Post-accident accessibility during severe accident is confirmed in PCSR Section 20.9. On the other hand, other requirements are considered in Safety Functional Claims, Safety Properties Claims, Human-Based Safety Claims and Human Factor Property Claims. These claims are referred in Appendix A and B of this chapter.

The Safety Category and Class for each Systems, Structures and Components (SSCs) are determined based on the approach described in PCSR Section 5.6. The following is the general rule for the safety categorisation and classification for the severe accident management measures.

- SSCs for severe accident management should be categorised as Category B because severe accident management measures make a significant contribution on nuclear safety and they are associated with the removal of radiological risks in the Beyond Design Basis Accident including Severe Accident.
- SSCs for severe accident management should be classified as Class 2 or Class 3. The principle or first-line means of delivering a Category B safety function is assigned as Class 2. A second-line means of delivering a Category B safety function is assigned as Class 3.

26.4.3.3 Basic Design of Severe Accident Management Measures

(1) Reactor depressurisation facility

The Reactor Depressurization Control Facility (RDCF) is designed to depressurise the RPV by opening safety relief valves in the event of design basis or beyond design basis faults including severe accident where SRV control from the Safety System Logic and Control (SSLC) are unavailable. The RDCF is designed to be operated manually from either the Backup Building (B/B) or the Main Control Room (MCR) in the Control Building (C/B). This facility is equipped with alternative power source (i.e. BBGs) and is equipped with a storage battery for safety relief valves and nitrogen supply equipment as transportable equipment. Considering the failure of the BBGs, the RDCF can also be activated by local manual operation in the reactor building, without any requirements for electrical power. The details about the RDCF are described in PCSR Section 16.7.3.3 (Reactor Depressurisation Control Facility).

(2) Core flooder system

Core flooder system is designed to inject water into the reactor core from the low pressure water supply system and prevent significant core damage and containment failure even if all emergency core cooling systems have failed. Low pressure water is supplied from permanent equipment (FLSS) or mobile equipment (FLSR). The design details of FLSS and FLSR is described in PCSR Section 16.7.3.1 (Flooder System of Specific Safety Facility) and Section 16.7.3.2 (Flooder System of Reactor Building).

(3) Containment heat removal facilities

Containment heat removal is attained by the recovery of RHR/LPFL system or by the use of the containment venting system.

(a) RHR/LPFL system can be recovered using the Diverse Additional Generator (DAG) or by large power truck when RHR/LPFL system has failed in the event of a “LOOP + Class 1 EDGs Failure”. The design details of the DAG and large power truck are described in PCSR Section 15.5 (Electrical Equipment and systems).
(b) RHR/LPFL system can be recovered using a Reserved Ultimate Heat Sink (RUHS) or an Alternate Heat Exchange Facility (AHEF) when RHR systems have failed in the event of a “Loss of Ultimate Heat Sink (LUHS)”. The detail design of the RUHS is described in PCSR Section 16.3 (Water Systems). The detail design of the AHEF is described in PCSR Section 16.7.3.6 (Alternate Heat Exchange Facility).
(c) The Containment venting system is designed to be actuated manually to prevent PCV damage due to overpressure. Containment venting is conducted using the Filtered Containment Venting System (FCVS) via the W/W because the most of fission products are expected to be scrubbed

into the suppression pool and the scrubber tank of the FCVS. In addition, considering the failure of remote manual venting operation, FCVS can be activated by local manual operation using extension valve from outside of the shielding wall, without any requirements for electrical power. Even if the manual PCV venting is failed, the containment venting is conducted passively because a rupture disk is installed in the FCVS. The detailed design of the FCVS is described in PCSR Section 16.7.3.5 (Filtered Containment Venting System).

(4) Containment Depressurisation facility

Containment depressurisation is conducted to prevent PCV failure due to overpressure. Containment depressurisation is attained using the same facilities as "(3) Containment heat removal facilities" described above.

(5) Containment Flooder system

The following systems are installed to provide the corium cooling in the lower D/W and provide containment cooling:

(a) Alternate D/W spray is designed to inject water into containment and control containment pressure and temperature to prevent/mitigate PCV failure due to overpressure and overtemperature. The FLSS or FLSR are used to conduct alternative D/W spray. The detailed design of the FLSS and FLSR is described in PCSR Section 16.7.3.1 (Flooder System of Specific Safety Facility) and Section 16.7.3.2 (Flooder System of Reactor Building).
(b) Lower D/W injection system is designed to inject water into lower D/W to mitigate MCCI in the case where the reactor vessel has failed and corium is discharged into lower D/W. FLSS or FLSR are used to conduct lower D/W water injection. The detailed design of FLSS and FLSR is described in PCSR Section 16.7.3.1 (Flooder System of Specific Safety Facility) and Section 16.7.3.2 (Flooder System of Reactor Building).
(c) Lower D/W flooder (LDF) is installed to passively supply water into the lower D/W to mitigate MCCI in the case where lower D/W water injection has failed. The detailed design of the LDF is described in PCSR Section 16.7.3.4 (Lower Drywell Flooder System).

(6) Hydrogen Control facilities

The following systems/measures are used to control hydrogen and oxygen concentration in the containment:

(a) Nitrogen inerting
In the UK ABWR, the containment is filled with nitrogen most of the time during at-power operation to ensure that oxygen concentration is controlled under the flammability limit even if flammable gases are generated due to metal-water reaction, Molten Core Concrete Interaction (MCCI) and radiolysis of water in the containment. Nitrogen gas is supplied from the AC

system through the High Pressure Nitrogen Gas Supply System (HPIN). The detailed design of HPIN is described in PCSR Section 16.2 (Process Auxiliary Systems).

(b) PARs in the containment

Passive Auto-catalytic Recombiners (PARs) are installed in the containment to recombine oxygen and hydrogen gases generated from the radiolysis of water. The detailed design of PAR is described in PCSR Section 13.3.3.3 (Primary Containment Vessel Gas Control System).

(c) PCV venting

In some of the accident scenarios where RHR/LPFL is not available, PCV venting is conducted to remove the decay heat from the containment and to prevent PCV damage due to overpressure. If PCV venting is conducted, hydrogen and oxygen are released to the environment and the PCV is filled with steam to maintain low combustible gas concentration. The detailed design of Filtered Containment Venting System (FCVS) is described in PCSR Section 16.7.3.5 (Filtered Containment Venting System).

(d) Alternative Nitrogen Injection system

After RHR/LPFL restoration, PCV venting line is closed and nitrogen is supplied from AC system or ANI (Alternative Nitrogen Injection) system to re-inert the containment and to prevent the PCV failure due to negative containment pressure differential. The detailed design of ANI is described in PCSR Section 16.7.3.7 (Alternative Nitrogen Injection System).

The following systems/measures are used to control hydrogen and oxygen concentration in the reactor building:

(a) PCV boundary

In the UK ABWR, PCV boundary is isolated in most of time during at-power operation. Even if large amount of hydrogen is released in the containment, the isolated containment limits the hydrogen release to the R/B.

(b) Standby Gas Treatment System (SGTS)

The SGTS is installed to maintain the secondary containment at a lower pressure than the environment and treat fission products that may leak from the PCV to the secondary containment before they are discharged to the environment. If AC power is available, SGTS can also be used to release the hydrogen from the R/B to the environment.

(c) PARs in R/B

A Passive Auto-catalytic Recombiner (PAR) is installed in the reactor building to recombine oxygen and hydrogen gases generated from the radiolysis of water.

(d) Blowout panel and entrance door for large equipment

If large amount of hydrogen is released to the reactor building as the result of a severe accident, the blowout panel and entrance door for large equipment are opened for hydrogen ventilation.

(7) Spent Fuel Pool Flooder System

The Spent fuel pool flooder system is designed to inject water into the spent fuel pool in order to cool the spent fuels and keep the water level high in the case where cooling function and injection function of spent fuel pool have failed and/or water leakage occurs in the spent fuel pool. This system has a function of not only water injection but also water spray. As the injection line to the SFP is connected to the spray headers which are located on the peripheral edge of the SFP, water injected to the SFP spreads across the entire surface area of the fuel racks and spent fuels are cooled by the spray when the SFP water level is low. The FLSS or FLSR are used to conduct SFP spray. The detailed design of FLSS and FLSR is described in PCSR Section 16.7.3.1 (Flooder System of Specific Safety Facility) and Section 16.7.3.2 (Flooder System of Reactor Building).

(8) Alternative Generator facility

The Back-up Building Generators System (BBGs) are installed in the backup building and are designed to supply alternative power source to severe accident management measures (e.g. FLSS pumps, SA C&I system) to prevent/mitigate core damage, containment failure, and fuel damage in the spent fuel pool in the event where Class 1 safety systems are not available. In addition to the above, even if RHR/LPFL systems are failed in the event of a “LOOP + Class 1 EDGs Failure”, one division of RHR/LPFL systems can be recovered by using Diverse Alternative Generator (DAG) or Large Power Truck. The details about the BBG, DAG, and Power Trucks are described in PCSR Section 15.5.4 (Backup Building Generator), Section 15.5.5 (Diverse Additional Generator), and Section 15.5.6 (Power Trucks).

(9) Severe Accident Instrumentation

The Severe Accident C&I System (SA C&I) is required to monitor the plant parameters which are necessary to understand the accident conditions and facilitate action to be taken to mitigate the consequences of a severe accident. Instrumentation is required to monitor the accident status of the core and spent fuel pool cooling, containment of radioactivity and radiation discharges. The SA C&I System also provides equipment to deliver the safety functions to control and to monitor the systems for severe accident management; this is mobile equipment that can be connected to the plant to provide means of mitigating the consequences of severe accidents. The detail design of SA C&I System is described in PCSR Section 14.6.6 (Severe Accident C&I System).

(10) Alternative Water Source

The alternative water source is designed to secure an adequate supply of water to fulfil the safety functional requirements to prevent/mitigate severe accidents. The alternative water source volume is designed to satisfy all required FLSS or FLSR injection functions for seven days, considering the following items.

- Amount of water supply for removal of the decay heat (core and lower D/W injection),
- Amount of water supply to the PCV spray,
- Amount of water supply to the reactor well, and
- Amount of water supply to the SFP.

The detailed design of FLSS and FLSR is described in PCSR Section 16.7.3.1 (Flooder System of Specific Safety Facility) and Section 16.7.3.2 (Flooder System of Reactor Building).

(11) pH Control

Alkali injection into suppression pool (S/P) is designed to control the pH of the S/P water and mitigate re-evaporation into containment gas phase of radioactive iodine retained in the S/P. The alkali injection system is designed to provide sufficient chemical injected volume and dispersion into the S/P to keep the pH near a neutral value or higher even in the event where acidic material is generated by radiation from fission products in the containment. Details on pH and iodine behaviour during transient in representative SA scenario and the effect of Alkali injection are described in [Ref-51]

(12) Other systems

A reactor well injection system is designed to inject water into the reactor well and to prevent the PCV damage in the drywell head flange due to overtemperature. This measure is based on the lessons learned from the Fukushima accident and is prepared to enhance the safety margin during a severe accident. The FLSS or FLSR are used to conduct reactor well injection. The detail designs of FLSS and FLSR are described in PCSR Section 16.7.3.1 (Flooder System of Specific Safety Facility) and Section 16.7.3.2 (Flooder System of Reactor Building).

26.4.3.4 Severe Accident Management

In the UK ABWR, various levels of accident management procedures will be prepared after GDA as follows:

- Abnormal Operating Procedures (AOPs),
- Emergency Operating Procedures (EOPs),
- Severe Accident Management Guidelines (SAMGs), and
- Emergency Operating Procedures for Plant Outage.

Figure 26.4-2 shows the overview of accident management guidelines for UK ABWR. AOPs are event-based procedures and used for postulated events that have been analysed and discussed in the Design Based Analyses, which is limited to single initiating events followed by successful operation of safety systems designed to respond to those events. EOPs are symptom-based procedures limited to before core damage and to respond to multiple failures induced severe accidents which are low frequency scenarios. SAMGs provide guidelines for mitigating accident scenarios in which severe core damage has occurred, and the failure of reactor pressure vessel and challenge to the containment integrity may follow. The severe accident analyses results presented in Sections 26.4.6, 26.4.7, and 26.4.8 provide the insights and information needed for the development of SAMG.

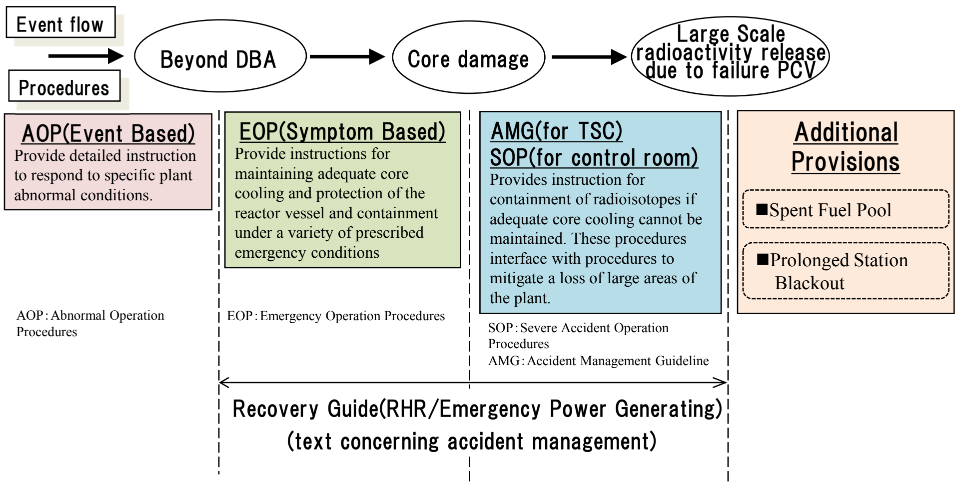


Figure 26.4-2 Overview of Accident Management Guidelines

26.4.4 Severe Accident Analysis Code

26.4.4.1 MAAP

Modular Accident Analysis Program (MAAP) is an integrated severe accident analysis code capable of modelling Fission Product (FP) release and thermal-hydraulic behaviour concurrently for the whole plant. It is a computer code that is widely used by nuclear utilities and research organisations to predict the progression of light water reactor accidents. The MAAP code is used as a severe accident code for the UK ABWR. Considering the applicable scope of the MAAP code for the UK ABWR, it is important to confirm that the MAAP code has physics models of events that could occur during an accident which leads to core damage and containment failure. These then must be incorporated and the models have to be properly validated by experiments and plant accidents. Therefore, MAAP physics models and MAAP validations were reviewed. The detailed model descriptions and benchmark results of the MAAP code are summarised in the MAAP user's manual [Ref-1].

In the UK ABWR, MAAP Version 4.07 is used for the severe accident analysis for at-power reactor operations and for the shutdown reactor. Hitachi-GE has significant experience using it to carry out severe accident analysis and it is well validated [Ref-37]. On the other hand, MAAP Version 5.03 possesses the modelling capability to perform severe accident analysis associated with the SFP regions of the auxiliary building. As with MAAP 4.07, similar validation has been conducted for MAAP 5.03. As a result, MAAP5 is used in the UK ABWR severe accident analysis for the Spent Fuel Pool (SFP).

(a) MAAP Physics model

Based on the scope of MAAP code, the incorporation of physics models into the MAAP code has been reviewed. In the results, it was found that the MAAP code has a variety of physics models that should be considered during severe accident. For example, the MAAP code incorporates the physics models of gas and water flow, natural circulation, steam evaporation and condensation, boiling, critical flow, conduction, convection and radiation heat transfer for level 1 phenomena. The MAAP code also incorporates the physics models of cladding oxidation and hydrogen evolution, core material eutectic formation, core relocation, lower head-core debris dynamics, failure of vessel penetrations and the lower head, debris entrainment, debris-concrete interactions, and fission product release, transport, and deposition [Ref-6].

The models for chemical behaviour in MAAP are supported by Operational Experience (OPEX), experimental data and theoretical study, etc. As iodine chemistry is not treated in the current severe accident analyses, it is independently studied and reflects on radioactivity release calculation. Following parameters are studied in detail [Ref-50],[Ref-51]

- Quantity and speciation of radioactivity release from fuel,
- Containment spray operation,
- Retention of radionuclides on containment wall surfaces,
- Dissolution and scrubbing of radionuclides into the Suppression Pool,
- Retention of radionuclides in the Suppression Pool,
- Retention of radioactivity in primary containment,
- Retention in secondary containment,
- Rate of leakage of radioactivity from Reactor Building,
- Retention and decay of radioactivity within Filtered Containment Venting System, and
- Iodine chemical behaviour in containment and suppression pool.

(b) MAAP validation

The review also showed that the MAAP physics models have been validated properly especially for BWRs. Through the benchmark of separate effect experiments or integral effect experiments, the

MAAP models related to thermal hydraulic behaviour of RPV, thermal hydraulic behaviour of PCV, damaged fuel behaviour, in-core fission product release, boiled-up liquid level, breaking-up of corium jet, and molten core concrete interaction has been benchmarked using a variety of experiments. Comparisons between the MAAP results and the experimental data show good agreement in all the benchmarks [Ref-6] [Ref-10]. As the benchmark for overall plant behaviour, the accident of Fukushima Dai-ichi nuclear power plant and Three Mile Island Unit 2 have been simulated by the MAAP code. As the results, the MAAP code provides a reasonable characterisation of the system response for not only the PWR plant [Ref-6] but also the BWR plant [Ref-7] [Ref-8] [Ref-9]. Therefore, it is concluded that the MAAP code has the ability to evaluate severe accident of the UK ABWR, which has similar design concept and similar nodalizations of BWRs.

26.4.4.2 JASMINE

The JASMINE (JAEA Simulator for Multiphase INteractions and Explosion) is one of codes used for the evaluation of ex-vessel FCI (steam explosion). Figure 26.4-3 shows analysis codes used for the evaluation of ex-vessel FCI. The JASMINE code is a mechanistic FCI simulation code developed by Japan Atomic Energy Agency [Ref-28]. The JASMINE code consists of two separate parts for the modelling of the molten core behaviour and the coolant multiphase flow. The molten core model includes three sub-models for the melt jet, melt pool and melt particles. The multiphase flow model, which handles the coolant thermo hydraulics, is a modified version of ACE3D code developed at Japan Atomic Energy Research Institute (JAERI) [Ref-27]. The melt jet and melt pool models are one-dimensional representations of a molten core stream falling into a water pool and a continuous melt body agglomerated on the bottom, respectively. The melt particles generated by the melt jet break-up in the water pool are modelled based on a Lagrangian grouped particle concept.

The process of steam explosion is modelled as follows:

- Premixing : Particle corium surrounded by vapour film,
- Triggering : Direct contact of water and the particle corium by vapour film break-up due to spontaneous or external disturbance,
- Propagation : Fragmentation and propagating, and
- Expansion : Energy transfer from thermal to mechanical.

For the verification of the explosion-related model functions, simulation of KROTOS experiments and FARO experiment at JRC Ispra of EU were performed and good correlations were obtained for KROTOS and FARO experiments. The models, correlations, verification and example calculations used in the JASMINE code are described in the JASMINE v.3 User's Guide [Ref-28].

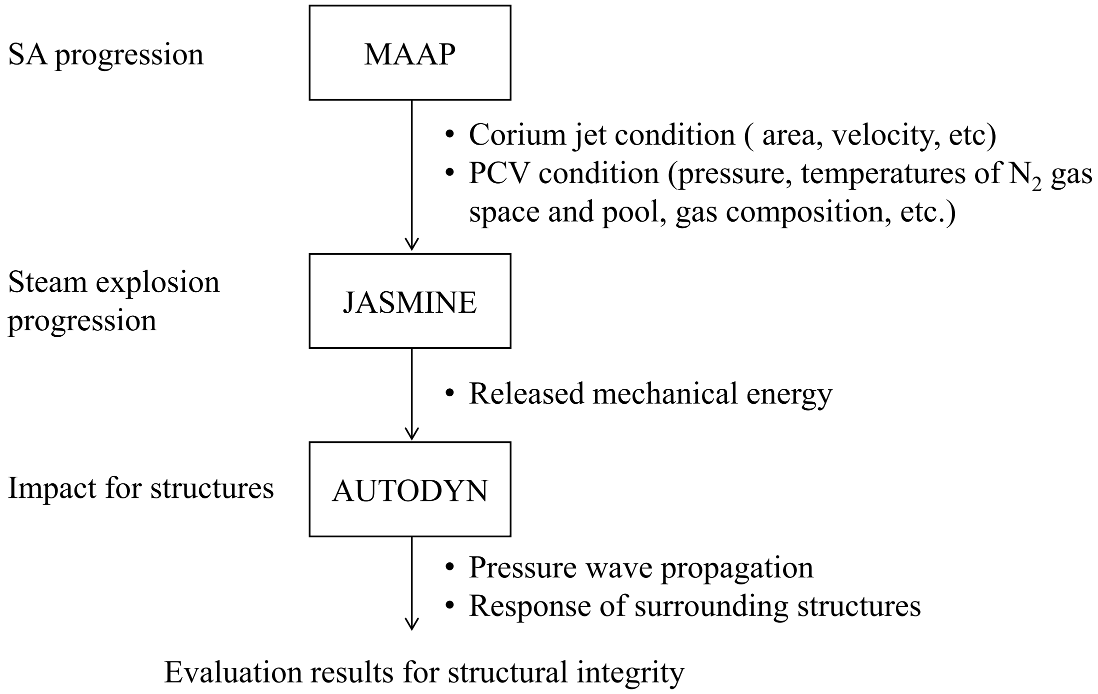


Figure 26.4-3 SA Analysis Codes used for the Evaluation of Ex-vessel FCI

26.4.4.3 AUTODYN

The AUTODYN is one of codes used for the evaluation of the structural response analysis following an ex-vessel FCI (steam explosion) as shown on Figure 26.4-3. AUTODYN code is one of the commercial Finite Element Method (FEM) codes which is appropriate to use for non-linear time history response analysis such as explosions and impacts [Ref-29]. Since this code is an adapted explicit method solver, which does not require convergence calculation and is different code from the finite element analysis tool adapted implicit method solver, it can be applied to evaluate the response to the following non-linear problems:

- Explosion (steam explosion), blast and shock wave propagation,
- Dynamics problems, including non-linear, and
- Fluid – structure interaction.

Many solvers are prepared as follows and many types of evaluation using gaseous, liquids and solids, etc., can be carried out. In addition, automatic and dynamic coupled analysis between solvers can be performed:

- Lagrange solver (FE),
- Euler solver (CFD),
- ALE solver, and

- Mesh free solver (SPH).

Analysis has been carried out as a benchmark for steam explosion, and the results demonstrate consistency with experimental results [Ref-30] [Ref-31] [Ref-32].

26.4.4.4 GOTHIC

GOTHIC (Generation of Thermal Hydraulic Information in Containments) is a versatile software package for transient thermal hydraulic analysis of multiphase systems in complex geometries. GOTHIC was developed by Zachry Nuclear Engineering Inc. (ZNE) under the sponsorship of Electric Power Research Institute (EPRI). It was developed within a QA program that complies with 10CFR50 and American Society of Mechanical Engineers (ASME) NQA-1[Ref-45] [Ref-46]. It is an internationally recognised computer code which is supported by a lot of validation exercises done by many organisations. GOTHIC solves the conservation equations for mass, momentum, and energy for multi-component, multiphase flow. A multi-block 3D grid with generalised flow connectors to connect the grid blocks is used to model complex geometries and systems. Many specialised features are included to model and control equipment typically found in ventilation and hydraulic systems (pumps, fans, valves, heat exchangers, etc.) A typical use of GOTHIC is to model nuclear reactor containment buildings. Hitachi-GE has used the GOTHIC code to assess the effectiveness of Hitachi-GE's hydrogen management in the Reactor Building. Before the application of GOTHIC for demonstration purposes, Hitachi-GE had confirmed the physics model of GOTHIC such as gas mixing, heat transfer, heat conduction and equipment model are supported by the appropriate benchmark [Ref-44].

26.4.5 PCV Failure Probability Evaluations

A review was performed to examine the extent of treatment needed in severe accident analysis to address the issue of phenomenological uncertainties and their impact on results. From the review, sensitivity analyses were defined for MAAP analysis. Three specific issues (i.e. ex-vessel FCI, DCH, and MCCI) relating to mechanisms leading to potential PCV failure were identified for more detailed uncertainty analysis in view of the complexity of the phenomenological issues involved. The analysis is based on the ROAAM (Risk Oriented Accident Analysis Methodology) approach [Ref-36]. This makes use of a probabilistic framework which allows the decomposition of a complicated phenomenon into a set of key constituent processes to be solved. The method has been applied in a number of US NRC issue resolution studies on key severe accident impact on nuclear plant risks [Ref-37]. A summary of the three evaluations is provided in PCSR Section 25.5.2.1 (Branch Probability for Phenomenological Event).

26.4.6 Severe Accident Analysis for Faults at Power

26.4.6.1 Severe Accident Analysis on Unmitigated Sequence

Severe accident analyses for unmitigated sequences were performed to evaluate the time margin for accident management such as equipment recovery and operator action.

(1) Analysis case and analysis conditions

In the Level 1 PSA, plant responses for various plant operations with the consideration of internal event, internal hazards (PCSR Chapter 7), and external hazards (PCSR Chapter 6) are modelled. The plant response paths are called accident sequences. Accidents sequences are categorised into twenty-six sequences in Level 1 PSA and they are taken over as Plant Damage State (PDS) into the Level 2 PSA. As some of sequences have almost the same accident progression as other sequences, twenty-one PDSs were selected as the unmitigated analyses cases. Table 26.4-2 shows the analysis cases for unmitigated sequence. The definition for each PDS is summarised in PCSR Section 25.5.1.2 (PDS Definition).

Table 26.4-3 shows summary of analysis conditions. Although the UK ABWR has adequate measures to prevent/mitigate the occurrence and propagation of the severe accidents as shown in Section 26.4.3.3, none of these measures are treated in this unmitigated analysis. Therefore, PCV failure is assumed to occur when PCV pressure or temperature reaches the ultimate capacity as shown in Figure 26.4-4. The evaluation of the severe accident analysis for unmitigated sequences was performed using the MAAP code as described in Section 26.4.4.1.

(2) Analysis results

The results of the severe accident analysis for unmitigated sequences provided the following key parameters, which were necessary for the development of containment event trees in Level 2 PSA (Section 25.5).

- Timing of core support plate failure,
  This is used to obtain the time margin available for RPV depressurisation and RPV water injection.
- Timing of RPV failure,
  This is used to obtain the time margin available for lower drywell injection for pre-flooding.
- Timing of PCV failure due to overpressure and over-temperature, and
  This is used to obtain the time margin for RHR initiation, RHR recovery, off-site power recovery, DG recovery, AC bus recovery, DC bus recovery and PCV venting.
- Timing of pedestal failure.
  This is used to obtain the time margin available for lower drywell injection for post-flooding.

Table 26.4-2: Analysis Case for Unmitigated Sequence

Case No.	Plant Damage State	Definition
1	S4	Excess LOCA inside containment
2	AE	Large LOCA with core injection failure
3	TQUV	High and low pressure core injection failure
4	TQUX	High pressure core injection and reactor depressurisation failure
5	TB	Station blackout
	TBU	Station blackout + RCIC failure, Same as TQUX
	TBP	Station blackout + failure to reclose SRV, Same as TQUV
	TBD	Station blackout + failure of DC power, Same as TQUX
6	TW	PCV heat removal failure followed by RPV damage at high pressure
7	TW-LP	PCV heat removal failure followed by RPV damage at low pressure
8	AW-LP	LOCA with failure of containment heat removal
9	AC	LOCA with failure of reactivity control
10	TC	PCV heat removal and reactivity control failure
11	TC-HP	Reactivity control and core injection failure, resulting in high pressure core damage in short term
12	TC-LP	Reactivity control and core injection failure, resulting in low pressure core damage in short term
13	S3E	ISLOCA or BOC with failure of RPV injection
14	S3UX	BOC (small liquid LOCA) with failure of high pressure injection and depressurisation
15	TNQUV	TQUV + SRV tailpipe break at W/W airspace
16	TNQUX	TQUX + SRV tailpipe break at W/W airspace
	TBPN	Loss of Class 1 and Class 2 AC power with SRV tailpipe break at W/W airspace, Same as TNQUV
	TBDN	Loss of Class 1 and Class 2 AC/DC power with SRV tailpipe break at W/W airspace, Same as TNQUX
17	TCN	TC + SRV tailpipe break at W/W airspace
18	AN	LOCA with failure of W/W to D/W vacuum breakers to open and failure of RPV injection
19	S12UX	Medium or Small LOCA + High pressure core injection and reactor depressurisation failure
20	S12W	Medium or Small LOCA + High and low pressure core injection failure
21	S12C	Medium or small LOCA + reactivity control failure

Table 26.4-3: Summary of 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	620 kPa[gauge]	Twice the maximum design pressure of PCV
Ultimate temperature of PCV	See Figure 26.4-4	Judged by PCV gas temperature
Decay heat power	ANSI/ANS-5.1-1979	MAAP model
Fuel type	GE-14	10×10 fuel rods, See PCSR Section 11.5
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
Failure area of PCV	0.068 m^2	Large failure area due to overpressure and overtemperature is assumed.

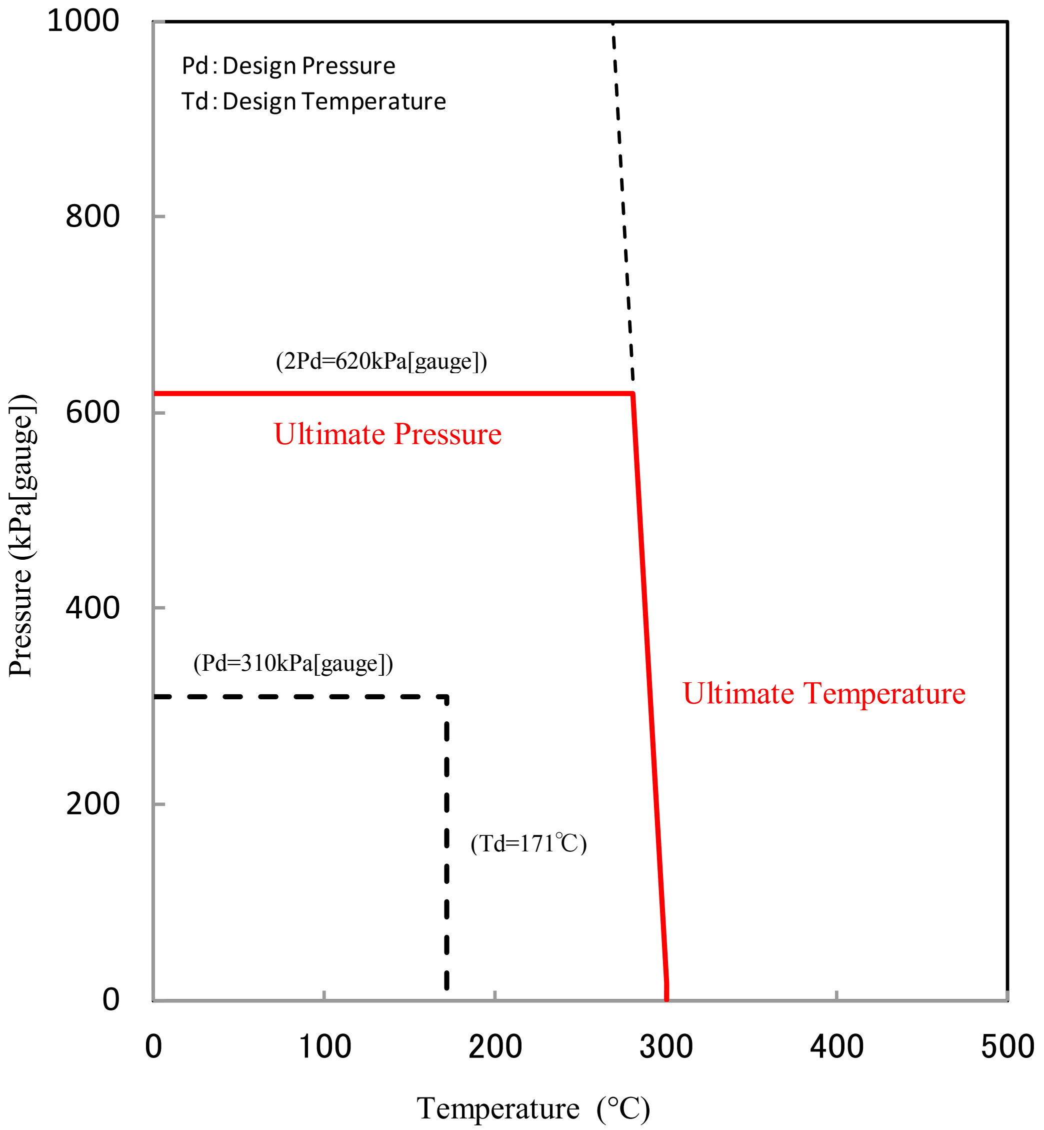


Figure 26.4-4: PCV failure criteria

26.4.6.2 Severe Accident Analysis on Mitigated Sequence

Severe accident analyses for mitigated sequences are performed in order to demonstrate the effectiveness of engineered features, strategies, and procedures during a severe accident. Mitigated severe accident analysis also supports the success path in the containment event trees and key event timing used in in Level 2 PSA.

(1) Analysis case and analysis conditions

The purpose of mitigated analysis is to show that containment failure can be prevented if some of the SA management measures are available in a severe accident. Representative analysis cases were selected from the success path of the containment event tree considered in Level 2 PSA. Table 26.4-4 shows analysis cases for mitigated sequences. The following SA management measures are considered in mitigated sequences:

- RPV depressurisation from RDCF,
- Alternative core injection from FLSS or FLSR,
- Lower drywell injection from FLSS or FLSR,
- Activation of LDF,
- Alternative drywell spray from FLSS or FLSR,
- Restoration of RHR, and
- PCV venting.

Mitigated analyses were conducted in the same analysis condition as the unmitigated analysis described in Table 26.4-3. The evaluation of the severe accident analysis on a mitigated sequence was performed using the MAAP code as described in Section 26.4.4.1. Relationship between system claims and severe accident assumptions is shown in Appendix D.

(2) Success criteria for effectiveness evaluation

Success criteria for effectiveness evaluation are shown in Table 26.4-1. The PCV failure criterion shown in Figure 26.4-4 is based on the containment performance capability assessment for the UK ABWR (See PCSR Section 10.6.4).

The containment performance capability assessment for the UK ABWR was done in three steps. Firstly, the accident condition which may lead to containment failure was identified. Secondly, the failure mode and locations were investigated based on the accident conditions. Thirdly, the failure conditions were evaluated for the failure locations and modes that were identified.

The following phenomena were investigated to demonstrate the requisite containment performance capability in the UK ABWR;

- Overpressure and over-temperature,
- Negative containment pressure differential,

- Hydrodynamic loading,
- Direct Debris interaction,
- Pedestal failure due to MCCI, FCI and DCH, and
- Other phenomena (e.g. subjecting metal components to extreme temperature gradients as a result of quenching).

The results of the investigation were used for the PCV failure criterion and for the modelling of the Level 2 PSA (see Section 25.5.2.3).

The containment performance of the UK ABWR is enhanced by the use of Ethylene Propylene Diene Methylene (EPDM) as the material for the flange gasket of the Drywell Head due to its higher heat resistance as opposed to the Silicone rubber that is used in Japanese BWRs [Ref-17]. The EPDM will also be used in the Equipment Hatch, Personnel Airlock and other small flanges. As the UK ABWR has various kinds of measures to mitigate the PCV temperature increasing (e.g. core injection, lower D/W injection, lower D/W flooder, PCV spray, and reactor well injection), the enhancement of the flange gasket material can provide an overall protection from containment leakage due to over-temperature. Furthermore, as the containment of the UK ABWR has a large capacity for overpressure and the Containment Overpressure Protection System (COPS) can limit the containment pressure increasing passively, the risk of PCV failure due to overpressure and overtemperature is decreased to as low as reasonably practicable in the UK ABWR.

(3) Analysis results

One of examples of the result for the mitigated analysis, Case 2-2 “Success scenario of Lower D/W injection + PCV Spray + PCV Venting in TQUV Sequence” of Table 26.4-4 is demonstrated in this section.

For the initiating event, the transient event with loss of feedwater is assumed. All high pressure core injection systems and all low pressure core injection systems are assumed to fail in this sequence. Although alternative core injection is assumed to fail, lower drywell injection, PCV spray, and PCV venting are assumed to be successful in this case.

Figures 26.4-5 to 26.4-6 show the analysis result. After loss of feedwater, the reactor water level decreases rapidly. Scram is initiated with a low reactor water level signal of L3, and MSIV is closed due to a low reactor water level signal of L1.5, but activation of high pressure injection systems (RCIC and HPCF) fails.

When the RPV water level drops below 20 percent of the active fuel length from the Bottom of Active Fuel (BAF), the RPV is assumed to be depressurised manually by opening two safety relief valves. However, as low pressure injection systems (LPFL and FLSS) are also assumed to fail, core damage occurs.

After that, the molten debris is relocated from the core to the lower plenum and the temperature of RPV lower head begins to increase. When the RPV lower head temperature reaches 300°C, operators start to inject water into the lower drywell and a water pool of 2 m is formed in the lower drywell.

RPV failure occurs and the corium is relocated into the lower drywell. As the corium is cooled in the lower drywell pool and steam is generated from the decay heat of the corium, the containment is pressurised. Operators begin to inject water into the lower drywell to continue the corium cooling.

When PCV pressure reaches 465 kPa[gauge] (1.5 Pd), D/W spray is performed intermittently and PCV pressure is controlled at less than 1.5 Pd. As the water level of the suppression pool gradually increases due to D/W spray and lower drywell injection, D/W spray is stopped so as not to submerge the W/W venting line when S/P water level reaches the elevation of V/B (Vacuum Breaker) -1 m height.

As a result, PCV pressure increases again and PCV venting from W/W is performed when the PCV pressure reaches 620 kPa[gauge] (2.0 Pd). PCV pressure decreases and the corium is cooled by lower drywell injection and PCV venting. The comparison between the analysis result and the success criteria is shown below.

(a) Overpressure and over-temperature

As shown in Figure 26.4-5 and 26.4-6, PCV pressure and temperature do not reach the ultimate capacity of the containment. Corium is cooled by water in the lower D/W as shown in Figure 26.4-8. The containment temperature in the upper drywell increases slightly after the PCV venting. However, if core injection or PCV spray is recovered after 72 hours, containment temperature decreases. Therefore, it is judged that the analysis satisfies the success criteria of overpressure and over-temperature.

(b) Direct Containment Heating (DCH)

As shown in Figure 26.4-7, the RPV can be depressurised to lower than 2.0 MPa [gauge] before RPV failure occurs by opening two safety relief valves at the time when the RPV water level drops below 20 percent of the active fuel length from the BAF. Therefore, it is judged that the analysis satisfies the success criteria of DCH.

(c) Hydrogen combustion

Figure 26.4-9 shows the gas concentration in the drywell and wetwell in Case 2-1 “Success scenario of core injection + PCV Spray + PCV Venting in TQUV Sequence” of Table 26.4-4. This is not the analysis result of Case 2-2, but this case can be referred to as the representative case for effectiveness evaluation for nitrogen inerting. As shown in the figures, oxygen concentration is the largest at the beginning of the accident, but it decreases due to the hydrogen and steam generation accompanied by severe accident progression. Furthermore, as PCV venting is conducted, oxygen and hydrogen are released to the environment and these concentrations decrease further. As the result, oxygen concentration does not reach the flammable limit of 5 percent. Therefore, it is judged that the UK ABWR satisfies the success criteria of hydrogen combustion as the containment is inerted with nitrogen.

(d) Ex-vessel Fuel Coolant Interaction (FCI)
Steam explosion evaluation was conducted using the JASMINE and AUTODYN codes and it is concluded that the supporting function of the reactor vessel is not lost even if steam explosion occurs in the lower drywell because the pedestal wall is made of thick concrete and steel structures [Ref-17]. In addition, Figure 26.4-5 shows that PCV pressure is less than the criteria for overpressure failure, 620 kPa[gauge], at the time of RPV failure. Therefore, it is judged that the success criteria for the prevention PCV failure following an ex-vessel FCI can be satisfied.

(e) Molten Core Concrete Interaction (MCCI)
As shown in Figure 26.4-8, the erosion of pedestal wall does not occur because pre-flooding in the lower drywell is conducted as an accident management action and the corium is easily quenched in the water pool. As there is a large uncertainty on the coolability of the molten debris, a sensitivity study has also been conducted and the results show that the success criteria for the prevention of PCV failure following an MCCI can be satisfied even if conservative analysis conditions (e.g. initial water level in the lower D/W, smaller spreading area of the corium, lower heat flux from the corium, and so on) are assumed [Ref-17]. Therefore, it is judged that the success criteria for the prevention of PCV failure following an MCCI can be satisfied when lower drywell injection or LDF are successful.
As shown the above, all of the success criteria are satisfied.

Table 26.4-4: Analysis Case for Mitigated Sequence

Case	PDS (Mitigation systems)	Sub-criticality *1	RPV				PCV					
			RCIC *2	Depressurisation	SRV tailpipe break in W/W airspace	Alternative Core 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	RHR Restoration	PCV venting
1-1	AE (Core+Spray+Vent)	O	X	-	-	O	-	-	-	O	X	O
1-2	AE (Lower+Spray+Vent)	O	X	-	-	X	O	O	-	O	X	O
2-1	TQUV (Core+Spray+Vent)	O	X	O	-	O	-	-	-	O	X	O
2-2	TQUV (Lower+Spray+Vent)	O	X	O	-	X	O	O	-	O	X	O
2-3	TQUV (LDF+COPS)	O	X	O	-	X	X	X	O	X	X	O
2-4	TQUV (LDF+RHR)	O	X	O	-	X	X	X	O	X	O	X
3-1	TQUX (Core+Spray+Vent)	O	X	O	-	O	-	-	-	O	X	O
3-2	TQUX (Lower+Spray+Vent)	O	X	X	-	X	O	O	-	O	X	O
4-1	TB (Core+Spray+Vent)	O	O	O	-	O	-	-	-	O	X	O
5-2	S12UX (Lower+Spray+Vent)	O	X	X	-	X	O	O	-	O	X	O
6-1	TW (Core+Spray+RHR)	O	O	O	-	O	-	-	-	O	O	-

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

*1: Reactivity control systems such as CRD and SLC are summarised in PCSR Section 12.4.

*2: Emergency Core Cooling System is summarised in PCSR Section 13.4.

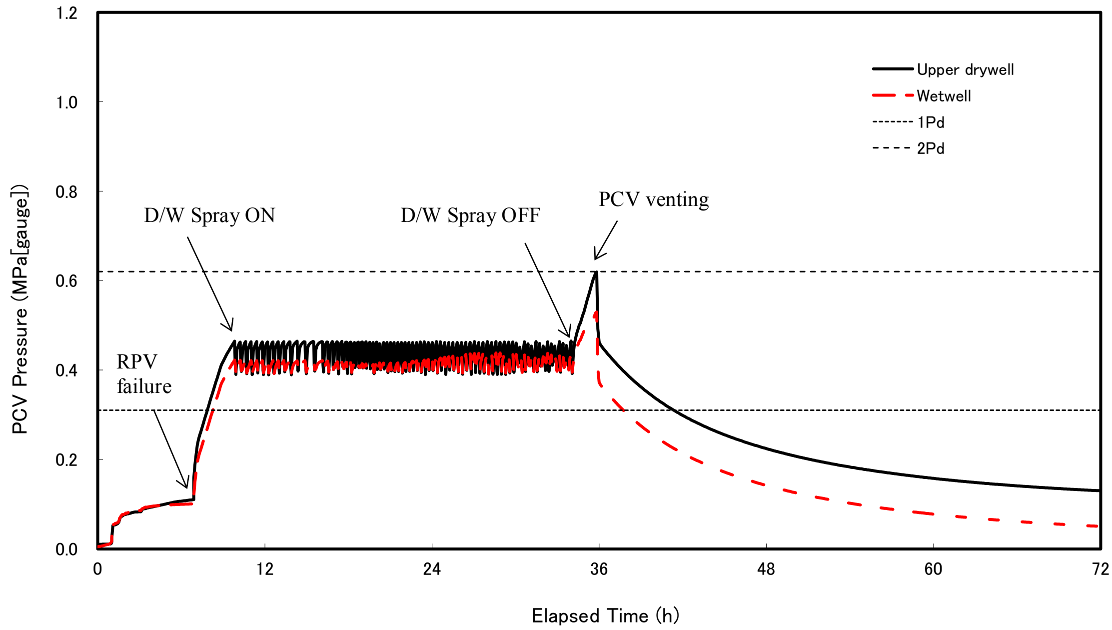


Figure 26.4-5: Containment Pressure in Mitigated Analysis Case 2-2

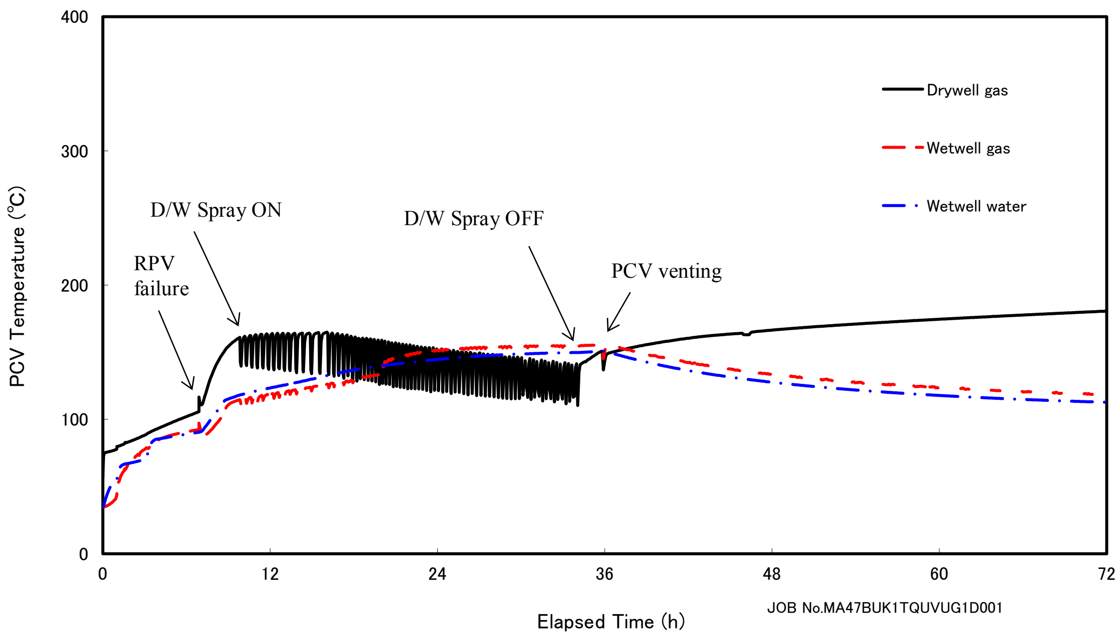


Figure 26.4-6: Containment Temperature in Mitigated Analysis Case 2-2

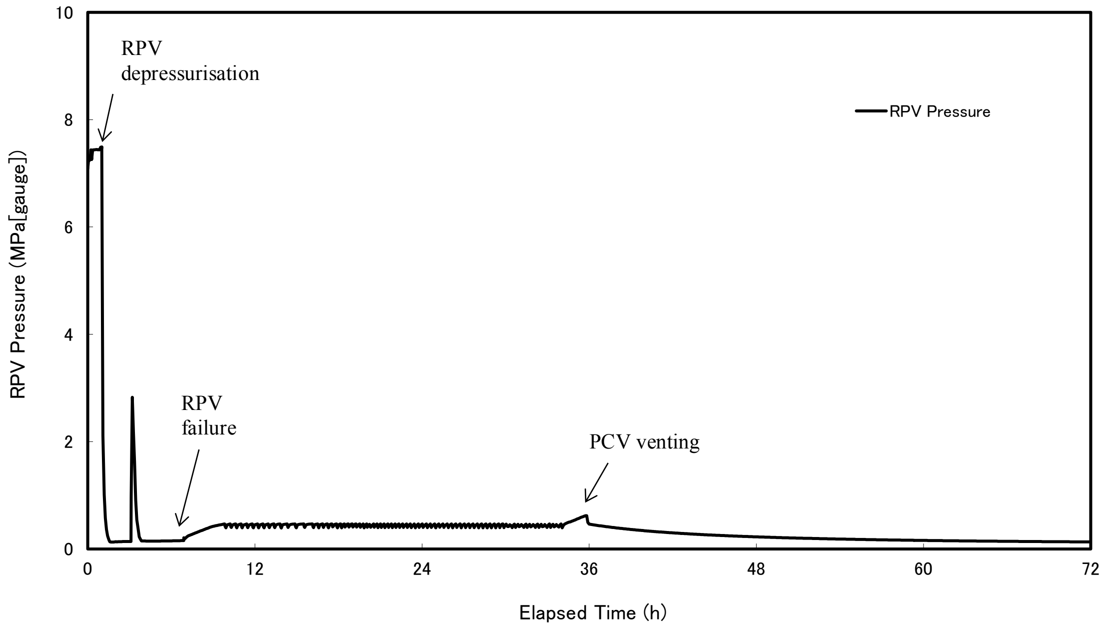


Figure 26.4-7: RPV Pressure in Mitigated Analysis Case 2-2

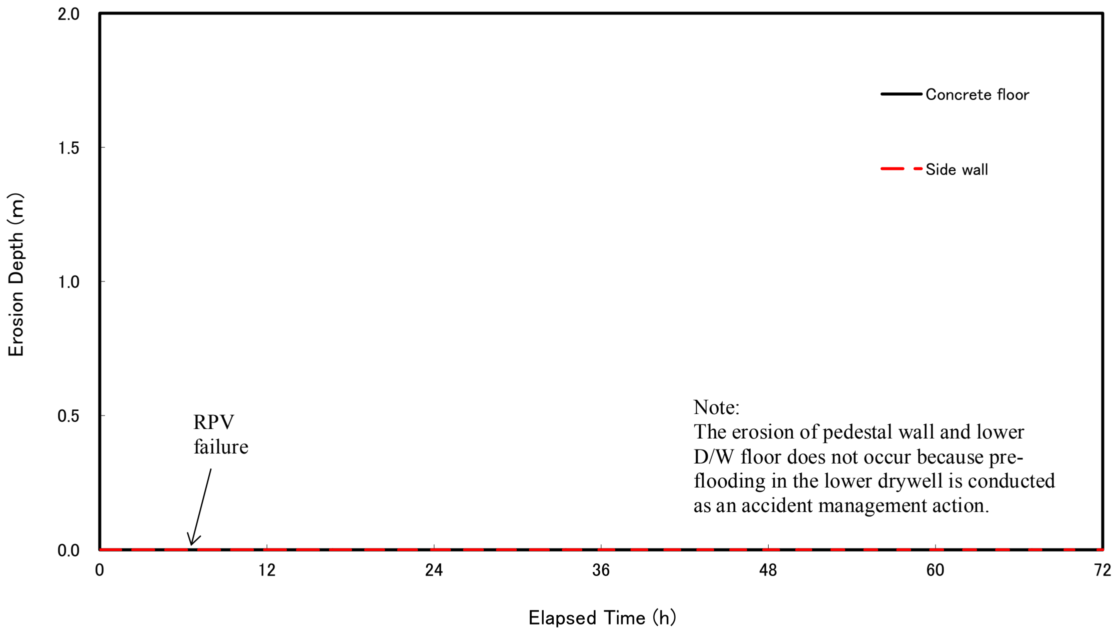
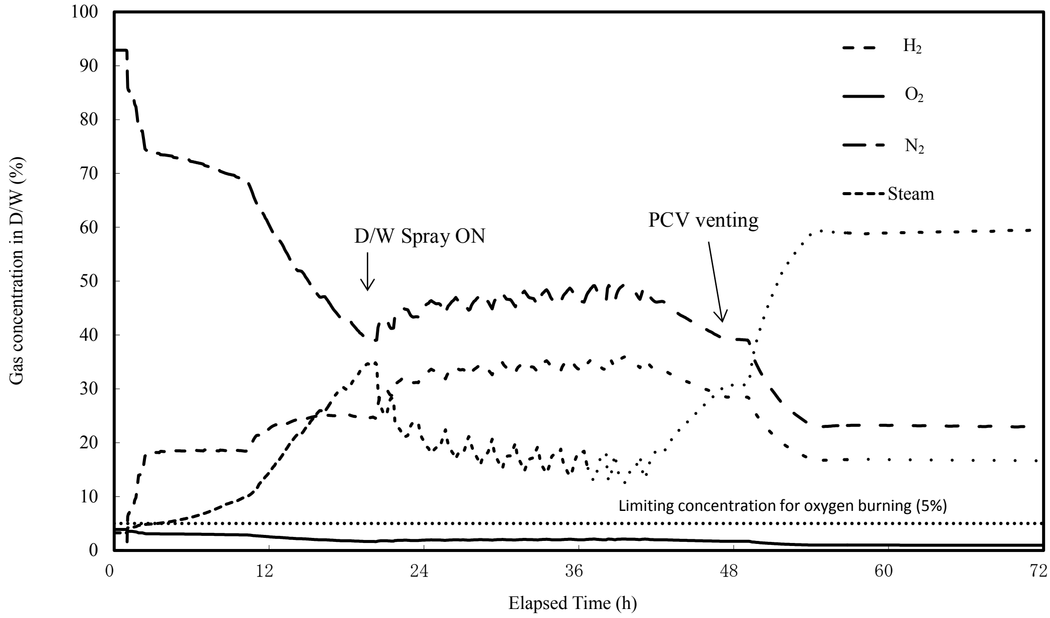
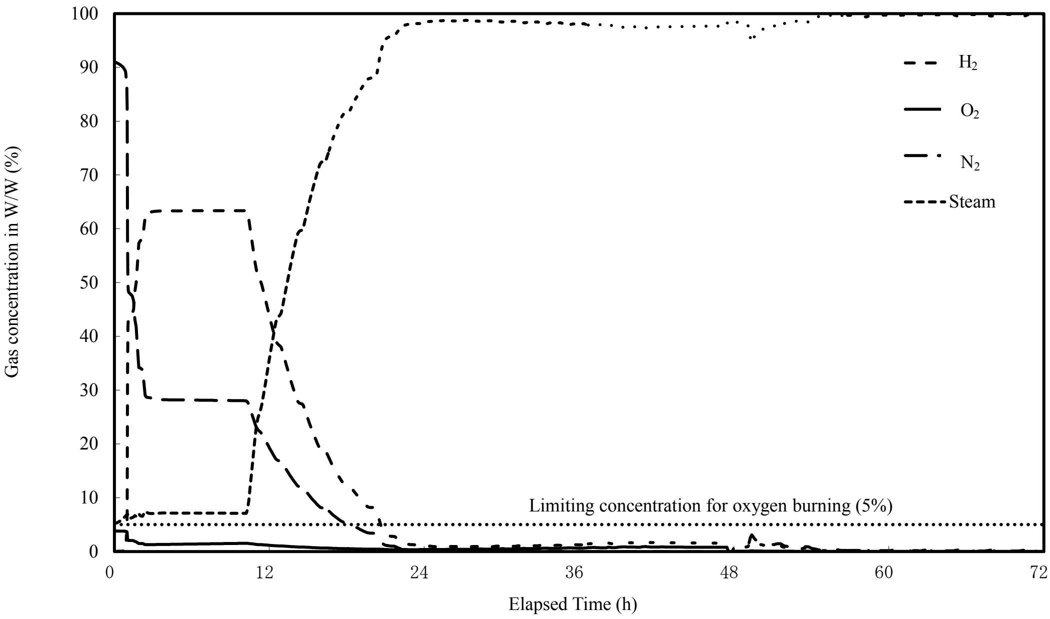


Figure 26.4-8: Erosion Depth in Mitigated Analysis Case 2-2



(a) Drywell



(b) Wetwell

Figure 26.4-9: Gas Concentration in Mitigated Analysis Case 2-1

26.4.6.3 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.
Nineteen end state identifiers are modelled in the containment event tree in the Level 2 PSA and 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. Release categories are defined to consider the magnitude and the timing of fission product release based on the characteristics of PDSs and the end states in containment event trees in the Level 2 PSA. The selection process of representative sequences for source term is described in PCSR Section 25.5.3 (Source Term Analysis). The following scenarios are considered as release categories in the source term analysis:

- Containment Leakage (with RPV Failure),
- Containment Venting (with RPV Failure),
- Filtered Containment Venting (with RPV Failure),
- Early Containment Failure,
- Late Containment Failure,
- Late Containment Failure (with PCV Spray Success),
- In-vessel Fuel-Coolant Interaction,
- Ex-vessel Fuel-Coolant Interaction,
- Direct Containment Heating,
- PCV Isolation Failure,
- Molten Core Concrete Interaction,
- RPV Rupture,
- Containment Bypass,
- S/P Bypass,
- Direct Debris Interaction, and
- Long-term SBO.

Source term analyses were conducted using almost the same analysis condition as the unmitigated analysis described in Table 26.4-3. A summary of the source term analysis is provided in PCSR Section 25.5.3 (Source Term Analysis).

26.4.7 Severe Accident Analysis for Faults at Shutdown

Some fault sequences initiated from reactor system faults on a shutdown reactor may result in severe accident conditions. A number of such fault sequences have been identified from the results of

Shutdown Level 2 PSA, which is summarised in PCSR Chapter 25.8 (Low Power and Shutdown PSA).

In the Shutdown Level 2 PSA, the following Plant Operating States (POS) were modelled for the reactor during a typical outage schedule during shutdown;

- POS S : Transition to a cold reactor shutdown,
- POS A : Transition to reactor disassembly and opening of the reactor well / pool gate with Division 2 in maintenance,
- POS B -1 : Full water level in the reactor well and gate open with Division 2 in maintenance,
- POS B-2 : Full water level in the reactor well and gate open with Divisions 1&3 in Maintenance,
- POS C : Transition to closed condition of the RPV and PCV heads with Divisions 1&3 in Maintenance, and
- POS D : Preparation of plant start-up.

The following scenarios have been determined as representative SA scenarios in the reactor during shutdown:

- Boil-off, RPV head open or No S/P Scrubbing, and
- Boil-off, RPV head closed.

Effectiveness evaluation for an alternative core injection as well as lower drywell injection (or LDF) has been conducted for the above scenarios. The summary of the effectiveness assessment is shown as follows:

- In both RPV head open and RPV head closed scenarios, effectiveness evaluation was conducted in the condition where alternative core injection is assumed to start after core support plate failure. The results showed that the RPV water level is recovered and stable cooling of damage core is attained.
- In both RPV head open and RPV head closed scenarios, effectiveness evaluation was conducted in the condition where LDF is assumed to activate after the RPV failure. The results showed that stable cooling of corium is attained in the lower drywell and fission products release to the environment is mitigated.

The UK ABWR has diverse and reliable core cooling measures available in its design to prevent core damage occurring. As such, the likelihood of core damage occurring is very low (see PCSR Section 25.8). Moreover, the possibility of core damage occurring during the short periods of time where the reactor is in shutdown mode is very unlikely because a large volume of water exists in the reactor well in the shutdown mode. Therefore, severe accident which leads to early or large fission product release in the shutdown reactor can be regarded as ‘practically eliminated’ [Ref-54].

Source term analysis has also been performed to determine the magnitude and characteristics of the source term used in the Level 3 PSA. A summary of the source term analysis is provided in PCSR Section 25.8 (Low Power and Shutdown PSA).

Relationship between system claims and severe accident assumptions is shown in Appendix D.

26.4.8 Severe Accident Analysis for SFP Faults

Severe accident plant states in the SFP can be identified from the result of the SFP Level 2 PSA, which is summarised in PCSR Chapter 25.9 (Spent Fuel Pool PSA). The design information on spent fuel storage facility, fuel pool cooling, clean-up and makeup systems, and spent fuel export systems, is summarised in PCSR Chapter Sections 19.8 to 19.10. The following scenarios have been determined as representative SA scenario in SFP:

- Boil-off,
- Small LOCA, and
- Catastrophic Failure of the SFP.

Effectiveness evaluation for the SFP spray has been conducted for the above scenarios. A summary of the effectiveness evaluations is as follows:

- In both Boil-off and Small LOCA scenarios, effectiveness evaluations were conducted for the condition where SFP spray starts to be performed 0.5 hours after fuel damage. As the result, SFP water level is recovered and stable cooling is attained. Fission product release to the environment is drastically reduced.

- In the Catastrophic failure scenario, effectiveness evaluation was conducted in the condition where SFP spray starts to be performed 0.5 hours after the accident initiates. When SFP spray is assumed to be uniformly distributed over the whole SFP area, fuel damage is prevented even four days after the reactor shutdown (corresponding to the decay heat power at the beginning of POS B and representing the highest decay heat power for hot fuel bundles) and there is no fission product release.

The UK ABWR has diverse and reliable core cooling measures available to prevent fuel damage in the SFP. As such, the likelihood of fuel damage is very low (see PCSR Section 25.9). In addition, there is a large volume of water present in the SFP which provides a large time margin for operators to restore SFP cooling if it is lost (Boil off and Small LOCA scenarios).

Source term analysis has also been performed to determine the magnitude and characteristics of the source term used in the Level 3 PSA. A summary of the source term analysis is provided in PCSR Section 25.9 (Spent Fuel Pool PSA).
Relationship between system claims and severe accident assumptions is shown in Appendix D.

26.4.9 Severe Accident Phenomenological Uncertainty Assessment

A comprehensive literature review from various sources was undertaken in order to identify the key severe accident phenomena and their associated uncertainties. This consisted of publications from the following sources;

- NUREG series [Ref-35] [Ref-34] [Ref-33] [Ref-25],
- Electric Power Research Institute (EPRI) [Ref-5], and
- MAAP4 application Guidance [Ref-26] .

The parameters identified from the literature review are provided in Table 26.4-5. Subsequently, sensitivity analyses and scoping studies were conducted for the parameters in order to confirm whether those parameters have significant impact on the off-site risk associated with the UK ABWR design. As severe accident analysis is performed to evaluate the time margin for accident management that is used in Level 1/2 PSA or to evaluate the magnitude and characteristics of the source term that is used in the Level 3 PSA, the impact on the time margin (accident progression) and the source term were investigated. The following is the conclusion of the sensitivity analyses:

- Impact on the Time Margin (Accident Progression)
  The results of evaluation conclude that timing of key event timing does not change significantly.
- Impact on the Source Term
  The results of evaluation conclude that amount of fission product release is not substantially changed but increases in some of the fission product groups. A conservative failure area is assumed in the base case of the severe accident analysis and the amount of fission product release decreases substantially if more a realistic failure area is assumed. Therefore, the impact on the risk due to uncertainty of model parameters is considered to be smaller than that due to the conservative failure area.

Table 26.4-5: Issues investigated in the UK ABWR sensitivity analysis

Number	Identified Parameters for Sensitivity Study
1	Core Melt Progression and Hydrogen Generation
2	Fission Product release from core
3	CsI re-evaporation
4	Time of vessel failure
5	Recriticality during in-vessel recovery
6	Direct containment heating
7	Ex-vessel FCI
8	Molten core concrete interaction
9	Containment failure location
10	Containment failure area
11	High temperature failure of drywell
12	Suppression Pool DFs
13	Main Steam Line creep rupture

26.5 Fukushima Accident Learning and Stress Test Assessment

The design of the UK ABWR has made use of the lessons learnt from the accident at Fukushima Dai-ichi and the outcomes of the European stress test reports to make provision for the flexible management of any external event beyond the design basis.

In the immediate aftermath of the Fukushima Dai-ichi accident which occurred in March 2011, the nuclear industry in the UK, along with the nuclear industries in many other parts of the world, undertook a review of existing and planned nuclear facilities to see if lessons could be learned from the accident that would improve safety. In the UK, licensees and designers based their responses on the following two main pieces of work:

- The recommendations made by the ONR Chief Inspector following his review of the accident.
- The ENSREG Stress Test specification, which was designed to explore the margins that each facility had in its design provisions for external events.

Hitachi-GE has gathered and listed all the issues raised by these exercises and items considered by licensees and designers:

- The recommendations of the ONR Chief Inspector,
- The responses by the licensees of selected existing UK nuclear reactor sites, (Hinkley Point C, Sizewell B, Torness, Wylfa)
- The responses of the Requesting Parties of UK EPR and AP1000 in the UK new build Generic Design Assessment programme,
- The review by the ONR of the UK responses to the ENSREG Stress Tests, and
- Observations and findings identified in a comprehensive report issued by the Director General of International Atomic Energy Agency (IAEA).

UK ABWR is compared against the considerations of Hinkley Point C and various operating nuclear power plants; Wylfa (existing station), Torness and Sizewell B. The rationale for the selection of operating UK plants is as follows:

- Sizewell B is representative of light water reactor design,
- Torness is representative of an operating Gas Cooled Reactors, and
- Wylfa is representative of Magnox reactors and it is the location for the first UK ABWR.

The current design of the UK ABWR has been compared to the findings of UK learning listed above [Ref-17]. In many cases, the early review performed by Hitachi-GE of the ABWR design in light of the Fukushima Dai-ichi event has led to similar and consistent conclusions and to similar design solutions. The main conclusions are:

- Against the issues identified that are applicable to GDA, the UK ABWR has design provisions already included in the current design or under design development to deal with the recommendations and Stress Test Findings.
- No omissions against relevant UK learning have been identified.

26.6 Assumptions, Limits and Conditions for Operation

26.6.1 Purpose

The purpose of characterising the Assumptions, Limits and Conditions for Operation (LCOs) during the GDA phase of the UK ABWR is to identify any constraints that must be applied by the future licensee to ensure safety during normal operation, fault and accident conditions. Some of these constraints are maximum or minimum limits on the values of system parameters, such as pressure or temperature, whilst others are conditional, such as the minimum level availability of a specified resource/equipment.

This section considers the links between the LCOs and the Beyond Design Basis / Severe Accident analysis that have been performed for the UK ABWR.

The LCOs that are applicable in this section are as follows:

- LCOs that guarantee the delivery of Safety Functions [SA and BDB]

The parameters that are identified in Section 26.3.5.2 are the LCOs for the BDB analysis. Therefore, in commercial operation, all these parameters are required to be kept within the defined range. If this is not achieved the deterministic safety case would be invalid. If any parameter goes outside the prescribed range then corrective action will need to be taken to correct this.

In Severe Accident analysis, the main aim is to demonstrate that the plant can be brought to and maintained in a stable condition after a severe accident. For example, this is achieved by ensuring the various severe accident phenomena presented in Table 26.4-1 are prevented by ensuring the associated success criteria conditions are met (i.e. ensuring the integrity of the containment) for an accident in an at-power reactor.

26.6.2 LCO to Guarantee the Delivery of Safety Functions

The beyond design basis analysis assumes the success or failure of a certain number of safety systems depending on the beyond design basis fault scenario that is being considered. As stated before, if the success criteria conditions presented in Section 26.3.3 are met, a beyond design basis fault can be controlled and mitigated. It has already been demonstrated that the UK ABWR can brought to and maintained in a stable condition during a beyond design basis fault in Section 26.3.5.

The following are the beyond design basis fault mechanical systems that deliver the required Safety Functions in a beyond design basis fault:

- Flooder System of Specific Safety (FLSS),
- Flooder System of Reactor Building (FLSR),
- Reactor Depressurisation Control Facility (RDCF),
- Filtered Containment Venting System (FCVS), and
- Containment Overpressure Protection System (COPS).

The design basis analysis described in Chapter 24 is performed assuming the plant is operated within a set of Limiting Conditions for Operation (LCOs) when a fault occurs and provides the justification of the values given. The analysis presented in this Chapter is performed on a best estimate basis and assumes initial conditions based on the nominal values associated with the LCOs discussed in Chapter 24. Consequently, the analysis presented in this Chapter does not contribute to the definition of the LCOs listed in Chapter 24.

The severe accident analysis assumes the success or failure of a certain number of safety systems depending on the severe accident scenario that is being considered. As stated before, if the success criteria conditions presented in Table 26.4-1 are met, a severe accident event can be controlled and mitigated. It has already been demonstrated that the UK ABWR can brought to and maintained in a stable condition during a severe accident in Section 26.4.6.2.

The followings are the severe accident management measures that deliver the required Safety Functions in a severe accident (see Section 14.6.6, 15.5.6, and 16.7):

- Flooder System of Specific Safety (FLSS),
- Flooder System of Reactor Building (FLSR),
- Filtered Containment Venting System (FCVS),
- Lower Drywell Flooder System (LDF),
- Alternate Heat Exchange Facility (AHEF),
- Reactor Depressurisation Control Facility (RDCF),
- Alternative Nitrogen Injection System (ANI),
- Power Supply Trucks, and
- SA C&I.

There are a number of systems that are provided solely for the purpose of Severe Accident management. These systems will be the subject of specific Severe Accident LCOs to ensure that when required to operate their availability and performance is consistent with the assumptions of the Severe Accident analysis. The detailed management of these Severe Accident LCOs will be specified post GDA.

Some of the LCOs for the severe accident management systems are specified in the Generic PCSR systems chapters (See Section 14.12 and 16.7).

26.6.3 Key Assumptions for Beyond Design Basis Analysis

The assumptions that were used within beyond design basis analysis in the UK ABWR GDA process and an explanation on the basis of the assumption are provided (e.g. scenario specific analysis) in [Ref-41].

The following is the different acceptance criteria from design basis analysis though the other acceptance criteria are same as design basis analysis in the beyond design basis analysis:

- Pressure on the reactor containment boundary is below the limiting pressure (i.e. 620kPa). Temperature on the reactor containment boundary is below the limiting temperature (i.e. 200 ºC).

26.6.4 Key Assumptions for Severe Accident Safety Case

A comprehensive list of assumptions that were used within severe accident analysis in the UK ABWR GDA process is provided in Appendix E of [Ref-17].

The following are key assumptions used in the severe accident analysis and safety case:

<u>Depressurisation</u>

- RPV is depressurised by opening two safety relief valves to prevent RPV failure with high pressure when RPV water level drops below 20 percent of active fuel length from the BAF.

<u>Water injection</u>

- Alternative core injection is conducted by FLSS or FLSR with the flowrate of more than 90 m^3/hr.
- Lower drywell injection is conducted by FLSS or FLSR with the flowrate of 90 m^3/hr when the RPV lower head temperature reaches 300 ^oC. Water level of 2 m in the lower drywell is formed before RPV failure to mitigate MCCI.
- Upper drywell spray is conducted by FLSS or FLSR with the flowrate of 300 m^3/hr when PCV pressure reaches 465 kPa[gauge] or PCV temperature reaches 200 ^oC. Upper drywell spray is turned off when the suppression pool water level reaches the elevation of V/B -1 m to prevent the wetwell venting line from submerging into the water.
- Reactor well injection is conducted by FLSS or FLSR with the flowrate of 70 m^3/hr when PCV temperature reaches 200 ^oC. Water level of 1.6 m is formed in the reactor well.
- SFP spray is capable of delivering 120 m^3/hr over the pool from FLSS or FLSR.
- FLSR is deployed in 8 hours after an accident.

- LDF activates passively when the lower drywell gas temperature reaches 260 ^{o}C. LDF is capable of providing water enough to remove decay heat of the corium.
- The small Power Truck can supply power to C&I and HVAC systems in the MCR and is deployed in 8 hours after an accident.

Decay heat removal

- AHEF can restore one division of RHRs in case of failure of RCW or RSW. AHEF Heat removal rate is 23 MW and is deployed in 35 hours after an accident.
- Wetwell venting through the FCVS is conducted by manually operators when PCV pressure reaches 620 kPa[gauge]. Even if manual venting fails, the rupture disk of the COPS bursts when PCV pressure reaches 620 kPa[gauge] and wetwell venting through the FCVS is conducted passively. Gas flow rate is 15.8 kg/s at PCV pressure of 310 kPa[gauge].
- The Large Power Truck can restore one division of RHRs in case of Station Blackout (SBO) and is deployed in 8 hours after an accident.

PCV capacity

- The ultimate capacity of the containment due to overpressure and overtemperature is 620 kPa[gauge] and about 300 ^{o}C.

Flammable gas control

- PARs in the containment keep the gas concentrations under the flammable limits, i.e. lower than 5 vol.% of oxygen or lower than 4 vol.% of hydrogen generated by radiolysis of water after a fault event without core damage.
- PARs in the R/B keep the flammable gas concentrations under the flammable limits, i.e. lower than 5 vol.% of oxygen or lower than 4 vol.% of hydrogen in case that the PCV boundary is intact.
- ANI can supply nitrogen gas into the containment with flowrate of more than 200 Nm^3/hr. ANI is deployed before RHR is recovered after PCV venting.
- The PCV boundary (equipment hatches and airlocks) is re-isolated in an accident during shutdown. It is not assumed in the severe accident analysis conservatively.
- If all of heat removal measures and water injection measures are failed, and re-isolation of the PCV boundary is failed in an accident of shutdown reactor or SFP, the blowout panel and the entrance door for large equipment is opened for the hydrogen management.

26.7 Summary of ALARP Justification

This section presents a high level overview of how the ALARP principle has been applied to the design of the mitigation systems for the beyond design basis accident and the severe accident, and how this contributes to the overall ALARP argument for the UK ABWR.

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. The approach to undertaking ALARP Assessment during GDA is described in the GDA ALARP Methodology [Ref-24] and Safety Case Development Manual [Ref-23].

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. This is discussed in Section 26.7.1.

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, including:

- A Backup Building with additional safety features to support core damage prevention and mitigation,
- Provision of mobile equipment, and
- Development of Accident Management procedures/Guidelines.

This section provides a summary of a number of evaluations to show that the design of the UK ABWR, in relation to severe accident mitigation, is consistent with Relevant Good Practice and the ALARP principle. In the GDA process, there were principally eight topics considered for ALARP evaluations in relation to severe accidents [Ref-43]. These are the following:

- 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. The first seven topics are discussed in Sections 26.7.2 to 26.7.8. Under the last item, the design for severe accident mitigation in UK ABWR is contrasted against a number of international guidance (IAEA, EPRI, EC) on what can be regarded as good practice on severe accident measures/strategies and this is provided in Section 26.7.9.

In addition, PSA insight for ALARP assessment and demonstration of practical elimination of large or early fission product release are provided in Section 26.7.10 and 26.7.11.

26.7.1 Beyond design basis accident

The BDB assessment presented in this chapter covers the full range of BDB risks but does not consider any additional nuclear safety risks for specific systems that are not already listed elsewhere in the PCSR.

Good practice in the design of systems that are important to the protection of the plant against BDB faults is considered elsewhere in the ALARP sections of the PCSR chapters that describe those systems. The methodology used for the BDB analysis itself follows good practice, as described elsewhere in this chapter, including such things as:

- Application of conservative assumptions such as the unavailability of all Class 1 SSCs.
- Acceptance criteria of fuel and reactor coolant pressure boundary for infrequent design basis fault are met with Class 2 & Class 3 SSC's.
- Correct performance of control systems is assumed if this exacerbates the BDB fault condition.
- Use of computer codes that have been validated for use for analysis of the types of BDB fault conditions that are being assessed.
- Application of acceptance criteria that are based on high quality experimental data and extensive worldwide operating experience.
- The analysis for each BDB fault group assessment considers the most onerous operating mode of the plant for that particular fault group.

The BDB section of the PCSR demonstrated that:

- There is considerable defense in depth and diversity in the UK ABWR design.
- The margins to the acceptance criteria are often significant.

As stated in Chapter 24 (Design Basis), the UK ABWR has been optimised where it is reasonably practicable to do so to further increase the margins to failure that are defined in the applicable acceptance criteria.

For example, the following provisions have been introduced in the UK ABWR design:

- The margin to the containment boundary limits can be improved with the adoption of a Diverse Additional Generator (DAG) which provides alternative AC power in some extreme Station Black-Out (SBO) scenarios (SBO is LOOP plus the failure of all EDGs).
- AHEF can be used in BDB fault conditions (and severe accidents) where the functions of RCW or RSW are lost in order to provide cooling water to one division of RHR Heat Exchangers and the associated auxiliaries to provide long term PCV heat removal.

In addition, other chapters in the PCSR demonstrate that there are no further reasonably practicable ways to further improve the provision of HLSFs claimed in the BDB assessment.

The conclusion is that as far BDB faults are concerned, there is no sudden increase in mitigated consequences as the frequency of events goes below the lowest DB frequency (no "CLIFF EDGE") and that the doses are ALARP.

26.7.2 Methods / technologies for confining an ex-vessel molten core

The UK ABWR design has diverse core cooling measures available to prevent and mitigate core damage and the potential subsequent relocation of corium from the RPV lower plenum to the lower drywell (see section 26.7.3). Even if core damage occurs, and the various cooling measures are successful, RPV failure is prevented and the corium is retained within the RPV.

However, there are still countermeasures available for confining an ex-vessel molten core in the UK ABWR design which includes the following features:

- A large spreading area in the lower drywell,
- Use of concrete with low potential for non-condensable and flammable gas generation,
- Water injection for debris cooling (FLSS, FLSR, and LDF),
- Countermeasures for PCV failure, and
- Countermeasures for PCV over pressurisation.

To demonstrate Relevant Good Practice, designs from previous ONR GDA applications and US NRC Design certification applications were examined [Ref-22] [Ref-21]. As a result, the following two design options were identified:

- The Basemat Internal Melt Arrest and Coolability Device (BiMAC) of the Economic Simplified Boiling Water Reactor (ESBWR) [Ref-22] [Ref-20] [Ref-19].
- The Core Melt Stabilisation System (CMSS) of the EPR [Ref-21].

An optioneering design study was carried out to consider the measures implemented in the ESBWR and the EPR for the confinement of an ex-vessel molten core [Ref-17]. This was assessed for the following key attributes:

- Spreading area,
- Floor type (material),
- Water injection measures, and
- Additional protection like refractory material or the BiMAC device.

It was demonstrated that the current measures in the UK ABWR for confining an ex-vessel molten core are consistent with Relevant Good Practice and no other reasonably practicable measures have been identified. The implementation of the BiMAC and CMSS concepts are not viable for adoption in the UK ABWR as the expected costs for such improvements are disproportionate in comparison to the benefits obtained.

Furthermore, analysis was carried out regarding the effect of ex-vessel FCI (steam explosion) on containment integrity (i.e. ensuring the confinement of an ex-vessel molten core) in a standard ABWR during a severe accident using the MAAP, JASMINE, and AUTODYN codes. The analysis demonstrated that, even if a steam explosion occurs, it does not lead to pedestal wall failure.

From the evaluation in [Ref-17], it can be concluded the measures available in the design of the UK ABWR for confining an ex-vessel molten core are appropriate and consistent with the principle of reducing risks to ALARP and no further reasonably practicable measures could be identified.

26.7.3 Methods of core or containment cooling

The UK ABWR core and containment cooling measures include the following features:

- ECCS (RHR/LPFL) : Primary measure , Active System,
- FLSS + Containment venting : Backup measure, Active system,
- LDF + COPS : Backup measure in SA, Passive system, and
- FLSR + Containment venting : Mobile backup measure, Active System.

As listed above, the UK ABWR has active measures, passive measures, and mobile measures available as a backup for the primary system (ECCS).

To demonstrate Relevant Good Practice, designs from previous ONR GDA applications and US NRC Design Certification applications were examined [Ref-18] [Ref-16]. As a result, the following design options were identified and examined:

- The Passive Safety Systems in the ESBWR [Ref-18], and

- The Passive Containment Cooling System in the AP1000 [Ref-16].

A design optioneering study was carried out for the core and containment cooling methods that are available in the UK ABWR in contrast to those that are available in the ESBWR and the AP1000 [Ref-17]. This was evaluated for the following systems:

- Water injection and containment heat removal system,
- Reactor well injection system, and
- Lower drywell injection system.

As a result, the following design options were adopted for the core and containment cooling in the UK ABWR:

- Water Injection and Containment Heat Removal System : ECCS (RHR/LPFL), FLSS + PCV vent, LDF + COPS, and FLSR + PCV vent,
- Reactor Well injection System : Post-flood operation after an accident, and
- Lower Drywell Injection System : Pre-flooding operation before RPV failure.

Furthermore, in the lessons learnt from the Fukushima accident, as reflected in Section 26.5, the design has additionally been enhanced by the use of mobile systems (FLSR, AHEF and mobile power supply trucks) and the inclusion of SAMGs, as reflected in Section 26.4. They can be regarded as reasonably practicable measures to further reduce risks.

From the evaluation in [Ref-17], it can be concluded that the core and containment cooling measures in the UK ABWR are appropriate as well being consistent with the ALARP principle and that no further reasonably practicable measures have been identified that could further reduce the risk.

26.7.4 Methods for further increasing grace / response times

The UK ABWR contains the following major systems to provide further increases in response / grace times:

- Core injection by RCIC,
- Alternative core injection by FLSS,
- Passive cooling of the corium by the LDF, and
- Passive containment venting by the COPS *[1].

To demonstrate relevant good practice, designs from previous ONR GDA applications and US NRC Design Certification applications were examined [Ref-18] [Ref-16]. As a result the following design options were identified:

- Passive Safety Systems in ESBWR [Ref-18] , and

*[1] Even if operators fail to conduct containment venting manually, containment venting is conducted passively by COPS. Then, operators can obtain the available time margin for other accident management procedures.

- Passive Containment Cooling System in AP1000 [Ref-16].

A design optioneering study was performed for these systems to examine the potential for further increasing the response/grace times in the design of the UK ABWR by considering those that are available in the ESBWR and the AP1000 [Ref-17].

As a result of the assessment, the following design options are adopted in the UK ABWR:

- Active and installed injection and heat removal system (RHR/LPFL),
- Active and installed injection system (RCIC),
- Active and installed injection and venting system (FLSS + Containment Venting),
- Passive injection and venting system (LDF + COPS), and
- Active and mobile injection and venting system (FLSR + Containment Venting).

From the evaluation in [Ref-17], it can be concluded that with regard to the measures available to further increase the response / grace times, there are no further reasonably practicable measures that could be identified that could further reduce the risks.

26.7.5 Methods of further capturing / reducing fission products inside containment

The UK ABWR has the following diverse methods for fission product capture/reduction;

- Suppression Pool (S/P) [Passive system] ,
- Containment Spray [Active System],
- Lower D/W injection systems [Active & Passive system],
- Standby Gas Treatment System (SGTS) [Active System],
- Filtered Containment Venting System (FCVS) [Active and Passive System],
- pH Control of the S/P [Active system].

To identify the relevant good practice, designs from previous ONR GDA applications and US NRC Design Certification applications were examined [Ref-18] [Ref-16]. As a result, the methods that are adopted and implemented for further capture / reduction of fission products in the containment were reviewed and summarised for the following reactors;

- ESBWR [Ref-18],
- AP1000 [Ref-16], and
- EPR [Ref-16].

A design optioneering study was carried out comparing the above reactor types against the UK ABWR with regards to the capture / reduction of fission products in the containment in a severe accident [Ref-17].

As a result of the assessment, it was shown that the fission products capture / reduction measures available for the UK ABWR, ESBWR, AP1000 and EPR are almost identical. However, the following measures for fission products capture / reduction unique to the designs are not reasonably practicable measures for consideration for the UK ABWR:

- The Passive Containment Cooling System (PCCS) of ESBWR,

- The Passive Containment Cooling System (PCS) of the AP1000, and
- The Annular Ventilation System (AVR) of the EPR.

It was shown that although the UK ABWR does not have these systems / measures available in its design, it has other comparable systems / measures that provide a similar function.

There has also been a separate design optioneering study conducted which demonstrated that the FCVS filter can reduce the releases of CsI and aerosols should venting occur in a severe accident condition. The FCVS Vent Filter is capable of DF $\geq$ 1000 for aerosols, and DF $\geq$ 100 for elemental iodine. In addition, it was shown the ALARP option during a severe accident for mitigating the release of gaseous iodine and the off-site impact is to conduct active pH control in the suppression pool.

Furthermore, the Level 3 PSA assessment for internal events at power indicated that the BSO for SP14.3.3 of the NSEDPs is met, reflecting the adequacy is the overall design of UK ABWR in the mitigation of fission product transport and releases to the environment. The FCVS function and pH control in suppression pool are considered as reasonably practicable measures that are evaluated for the design, although the benefits through Level 3 PSA have been shown to be very marginal.

From the above evaluations, it can be concluded that the UK ABWR design is consistent with the ALARP principle in regard to the available methods for further reduction / capture of fission products and no additional reasonably practicable measures could be identified.

26.7.6 Design of the PCV head flange and other systems to limit PCV leakage

The UK ABWR has the following countermeasures available for containment leakage:

- Debris and Containment Cooling (FLSS, FLSR, and LDF),
- Enhancement of the heat resistant material used for the gasket compared to Japanese BWRs,
- Drywell Head cooling, and
- Containment Venting (FCVS and Hardened Containment Venting System).

The following relevant good practices that are adopted internationally in other nuclear power plants as means of providing containment leakage protection were identified and investigated:

- Core and containment cooling measures,
- Heat resistance material for the Gasket, and
- Drywell Head Cooling.

Considering the identified relevant good practices, a design optioneering study was carried outlining how the UK ABWR represents relevant good practice and follows the ALARP principle with regard to containment leakage prevention and mitigation [Ref-17]. As a result, the following design options are adopted in the UK ABWR:

- Core and containment cooling measures (See section 26.7.3),
- EPDM for the Drywell head flange gasket material, Equipment Hatch, Personal Airlock, and small penetrations, and
- Reactor well injection during an accident.

Furthermore, an effectiveness evaluation of the reactor well injection system is provided to demonstrate that PCV head failure can be prevented under severe accident conditions [Ref-17].

From the evaluation in [Ref-17], it can be concluded that the UK ABWR is consistent with Relevant Good Practice and the ALARP principle with regard to containment leakage prevention and mitigation and no further reasonably practicable measures have been identified.

26.7.7 Methods for flammable gas control

The reference ABWR design contains the following systems for flammable gas control in the containment and the reactor building:

- Nitrogen Inerting [Containment],
- Flammability Control System (FCS) [Containment],
- Standby Gas Treatment System [Reactor building], and
- Blowout panel [Reactor Building].

To identify relevant good practice, flammable gas control systems / measures that are used internationally were investigated for various reactor types (BWRs, PWRs, VVERs and PHWRs). As a result, the following systems were identified as Relevant Good Practice for flammable gas control but are not available in the reference ABWR design:

- Igniter
- Passive Auto catalytic Recombiner (PAR)

The design options for the containment and the reactor building were then assessed, considering the above flammable gas control systems [Ref-17]. After considering the advantages and disadvantages of each design option, the following flammable gas control measures were determined to be adopted for the UK ABWR:

Flammable gas control measure for the containment:

- Early stage of an accident : Nitrogen Inerting,
- Middle stage of an accident (before the restoration of RHR) : PCV venting, and
- Late stage of an accident (after the restoration of RHR) : RHR + PAR,
- Final stage of an accident (temporarily used to remove hydrogen gas) : RHR + Nitrogen injection

Flammable gas control measure for the reactor building:

- Small amount of hydrogen leakage to the reactor building : SGTS,
- Further reduction of hydrogen concentration : PAR, and

- Large amount of hydrogen leakage to the reactor building : R/B Opening with Defensive Shield.

Effectiveness evaluation is conducted for the following:

- PAR (both containment & R/B),
- SGTS, and
- openings in the R/B.

The analyses confirmed that the concentration of flammable gas in the containment and the R/B are kept below the flammable limit in a severe accident when the reactor is at-power.

In addition to the above, flammable gas control measures for an accident during shutdown or in SFP are also discussed considering the advantages and disadvantages of each design option. The following were determined to be adopted for the UK ABWR [Ref-17].

- Enhancement of water injection by optimisation of plant outage schedule.
- Enhancement of water injection by standby of FLSR.
- PCV Isolation by closing PCV Hatches and Airlocks after an Accident.
- Passive Auto-catalytic Recombiner (PAR).
- Standby Gas Treatment System (SGTS).
- R/B Venting through Blowout Panel.

Generally, as the large volume of water present above the fuel in the reactor well during shutdown or in the SFP, there is a large time margin available to prevent fuel damage. Therefore, most of accident in the shutdown reactor and in the SFP can be terminated without fuel damage and no hydrogen is generated.

From the evaluation in [Ref-17], it can be concluded that the UK ABWR is consistent with Relevant Good Practice and the ALARP principle with regard to the methods used for flammable gas control and no further reasonably practicable measures have been identified.

26.7.8 Containment venting

A design optioneering study was carried out for the containment heat removal systems considering the following design options which are adopted in the reference ABWR design [Ref-17].

- RHR system,
- Containment venting,
- AHEF,
- Passing water through D/W cooler,
- Alternative Heat Removal with CUW (Reactor Water Clean-up System), and

- Alternative RHR (Direct S/P cooling) system.

After considering the advantages and disadvantages of each design option, it was determined that the following measures were adopted for containment heat removal in the UK ABWR;

- RHR system: Primary measure,
- Containment venting : Secondary measure,
- AHEF : Backup to recover the RHR system,
- Passing water through D/W cooler : Further mitigation measure, and
- Alternative heat removal with Reactor Water Clean-up System (CUW) : Further mitigation measure.

As can be seen above, containment venting is the backup measure for containment heat removal in the UK ABWR. The RHR system can be recovered using the RUHS or the AHEF in the event of a "Loss of Ultimate Heat Sink (LUHS)". In addition, the RHR system can be recovered using DAG or large power truck in the event of a "LOOP + Class 1 EDGs Failure". On the other hand, the containment venting system has the following advantages:

- It consists of piping, isolation valves and a control system. It is simple system and is able to withstand the effect of an extreme severe event.
- Operator action and power for active components are not required as COPS is installed.

Containment venting has the disadvantage of releasing fission products to the environment after PCV venting. However, this can be mitigated by the FCVS. A large part of the fission products released from the fuel is scrubbed in the S/P and the vent filter of the FCVS. Furthermore, it was demonstrated the risk does not become large as a result of containment venting. In addition, if the pH of the suppression pool is properly controlled, the risk of PCV venting is decreased further. From the evaluation in [Ref-17], it can be concluded the UK ABWR is consistent with the ALARP principle with regards to its containment heat removal systems and no further reasonably practicable measures have been identified.

26.7.9 Additional severe accident management measures

The evaluation on how the design of the UK ABWR is consistent with Relevant Good Practice and the ALARP principle with regard to additional SA measures has been undertaken. A summary of this evaluation is described below. A summary review of the SA measures for water injection, heat removal, depressurisation by relief valves or vent line, reactivity control, fission product removal, and prevention of hydrogen combustion in order to prevent severe accident and to mitigate its progression is presented within the supporting document [Ref-17]. As a result, the need to outline the ALARP discussion used to justify the approach taken for the following two items was identified:

- Severe accident monitoring system, and
- Measures to depressurise the RPV.

In regard to severe accident monitoring systems, it has been justified that monitoring parameters for severe accident management are properly identified based on the Severe Accident Management Guidelines (SAMGs). Monitoring parameters can be divided into three categories; Primary parameters for severe accident management, Backup parameters for severe accident management, and Specific parameters for severe accident management. SA C&I system will be prepared and qualified based on the categories (importance) for each parameters. The detail is described in PCSR Section 14.6.6 (Severe Accident C&I System). As for RPV depressurisation measures, adequacy of SRV design has been justified [Ref-48]. SRVs consist of sixteen independent valves so that their reliability is very high.

The international good practice on SA measures / strategies that are cited within literature for Severe Accident Management (SAM) were also investigated and summarised. Documents of the IAEA, EPRI, and the European Commission (EC) [Ref-13] [Ref-11] [Ref-4] were selected as representative international good practice. The SA measures / strategies in the three documents that should be considered for SAM were compared against the current SA measures available in the design of the UK ABWR. It was shown that the UK ABWR considers most of the relevant measures / strategies identified within the three documents. When a relevant SA measure / strategy were identified that is not considered for the UK ABWR, an explanation was provided as to why this was the case.
From the evaluation in [Ref-17], it can be concluded that the SA measures / strategy in the UK ABWR is consistent with Relevant Good Practice and the ALARP principle and no additional reasonably practicable measures have been identified that could further reduce the risks.

26.7.10 PSA Insight for ALARP Assessment

The PSA has been developed in order to demonstrate that the risk associated with the design and operation of the UK ABWR is ALARP and that it can be concluded that there are no further reasonable practicable improvements identified for the generic plant design during GDA [Ref-49].

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 are Internal Events at Power, Internal Events Shutdown, Internal Events SFP, Internal Fire at Power, Internal Flood at Power, Seismic, Fuel Route and Other hazards. The general process can be described in two steps:

- Step 1:
  A systematic review of each fault group to be performed in support of the ALARP demonstration. Risk significant characteristics will be summarised from the PSA results. This includes a systematic review identifying any plant vulnerabilities and major modelling uncertainties in the PSA or other improvements that could be made in the plant design or operation to reduce the plant risk.
- Step 2:
  PSA insights affecting the ALARP determination that risks are ALARP have been established for risk significant characteristics. The results from each of the faults have been reviewed systematically to determine if improvements could be made to the design or operation of the facility to make the risks as low as reasonably practicable.

As the result of review, 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-49]. Table 25.15.5-1 to Table 25.15.5-8 in PCSR Section 25.15 (Summary of PSA Results and Key Insights) show the PSA key insights from each PSA. The following are the PSA key insights related to beyond design basis accident and severe accident.

The following is an example of the key insights related to the BDB analysis.

- CCF of SRVs (Failure to open all SRVs)
  There has been a study undertaken which considered the potential options available to further enhance the protection in the UK ABWR against CCF of all the SRVs [Ref-48]. The potential of installing additional discharge lines on the Main Steam pipe controlled by motor operated valves, diversity in the selection of SRV manufacturers, improvement in the maintenance work etc. to prevent CCF of all the SRVs were investigated.

However, in the unlikely scenario where all means to open the SRVs mechanically fail due to CCF, this would lead to a very infrequent LOCA for which there are multiple and diverse mitigation systems available. The SRVs adopted for the UK ABWR has been shown to be reliably operated for 30 years without any incidents in comparison to other less reliable SRVs that were adopted in some of the earlier BWRs (Pilot type SRVs). Moreover, the contribution this has on the total CDF in the PSA is insignificant compared to other events (see PCSR Chapter 25 15.5).

From the above discussion the identified improvements are disproportionate to the benefits derived and the current SRV design at a system level is considered to be ALARP.

The following are the key PSA insights related to SAA.

- PSA-IR-0006: Protection of Hatches in Access Tunnel
  In the Level 2 PSA, the hatches are assumed to fail with 1.0 probability given RPV failure at high pressure. An alternative option is to introduce an additional hatch inside the existing hatch for both lower drywell tunnels.
- PSA-IR-0009: COPS Setpoint
  COPS is included as part of Filtered Containment Venting System (FCVS), and provides protection against the event of containment over pressurisation. A rupture disk opening at twice the pressure of PCV design pressure (2Pd) is mounted on downstream of PCV isolation valves from W/W. On the other hand, HPCF pump is assumed to be terminated at 150^{o}C of S/P water temperature. If COPS setpoint is lower than 2Pd, HPCF pump can continue to run.

As the result of the review for the above identified items, no further reasonably practical measures are identified. Therefore, it concludes that the UK ABWR design in GDA complies with the principle that risks are ALARP, and that for GDA no further reasonably practical measures to reduce risks need be considered.

26.7.11 Practical Elimination

The demand for continuous improvement in safety of nuclear power plants has led to a requirement for new nuclear power plant designs to provide a demonstration of ‘Practical Elimination of Early or Large Fission Product Release’ as a result of severe accidents [Ref-55]. Based on a review of international guidance (IAEA, WENRA and EUR) and the identification of key attributes/requirements necessary for such demonstration, a framework comprising of 3 steps is developed for this evaluation. The three steps are:

- Step 1 Design provisions,

- Step 2 Identification of ‘conditions’ to be considered for demonstration of elimination, and
- Step 3 Demonstration of ‘extremely low likelihood’ with a high degree of confidence.

In Step 3, the following numerical targets are proposed for the demonstration of practical elimination for the UK ABWR:

- Large Release frequency of 10^{-6} per reactor year,
- A value of 10^{-7} per reactor year for individual RCs,
- BSOs for Targets 8, and
- BSOs for Targets 9.

Overall, against 4 numerical targets defined for the demonstration framework, it can be concluded that the UK ABWR has broadly met the principle of ‘practically elimination of large or early releases of radioactivity in the event of a severe accident’ [Ref-54]. This has been demonstrated for internal initiators for reactor at-power, shutdown and the SFP. The basis of the demonstration is principally based on a combination of consideration of the diverse design measures and defence in depth provisions in the design and supported by results from extensive PSA and severe accident analysis. A more conclusive demonstration can be achieved post GDA when more detailed design information and operator procedures are available.

26.7.12 Summary

For all BDB faults analysed, there is no sudden increase in mitigated consequences as the frequency of events goes below the lowest DB frequency (no “CLIFF EDGE”) and that the risks are ALARP.

The reference ABWR design is equipped with extensive design features for the mitigation of severe accidents. Consistent with the principle of ALARP, the following should be noted:

- The UK ABWR design has been further reinforced to improve the resilience against severe accidents following the Fukushima Dai-ichi accident, including the following design enhancements:
  - A Backup Building with additional safety features to support core damage prevention and mitigation:
  - Provision of mobile equipment (i.e. FLSR, AHEF, power trucks, and ANI)
  - Development of Accident Management procedures/Guidelines.
- A comparison of the SA measures / strategies included in the UK ABWR has been made against international guidance (IAEA, EPRI, and EC) on candidate designs/actions essential for severe accident mitigation. The comparison showed that the demands have been met so far as is reasonably practicable in the UK ABWR design.

- An assessment has also been made for 7 specific severe accident management strategies against other options. The assessment confirmed that the UK ABWR design features are consistent with Relevant Good Practice and no reasonably practicable improvements have been identified for further considerations.
- As the result of the review for the PSA insight, no further reasonably practical measures for severe accidents are identified.
- From the review of international guidance (IAEA, WENRA and EUR) and numerical targets proposed for the demonstration of practical elimination, it is concluded that the UK ABWR has broadly met the principle of ‘practically elimination of large or early releases of radioactivity in the event of a severe accident’.

26.8 Conclusions

This chapter provides a summary of the assessment of Beyond Design Basis (BDB) Faults and Severe Accidents (SA).

Beyond Design Basis Faults

BDB Analysis identifies and analyses credible fault sequences that have lower frequencies than those included in the Design Basis, typically as the result of assuming multiple failures or common cause failures of protection systems following an initiating event. Such fault sequences could potentially result in significant radiological consequences. The aim of the BDB analysis is to show that such low frequency fault sequences do not contribute disproportionally to nuclear risk i.e. that there is no sudden increase in the severity of the radiological consequences as the Design Basis boundary is crossed. The analysis is comprised of the following steps:

- Identification and grouping of BDB faults,
- Identification of relevant Acceptance Criteria from DB fault studies,
- Identification of SSCs that provide protection against BDB faults, and
- Transient analysis of the identified events to show Acceptance Criteria are met.

The scope of analysis has included the following:

- BDB faults at power,
- BDB faults in shutdown modes, and
- BDB faults in SFP and fuel route.

Some key conclusions can be drawn from this analysis:

- The acceptance criteria are met and often with considerable margin.
- No BDB faults in the GDA fault schedule lead to either considerable damage or melting of the core, so that no significant environmental release of any radioactive material occurs.
- There is considerable defence in depth and diversity in the UK ABWR design to deal with these faults.

Severe Accidents

The UK ABWR design is equipped with extensive design features for the prevention and mitigation of severe accidents. In the highly unlikely event of a severe accident, the design will ensure that any potential environmental release of any radioactive material from the plant during all modes of operation is acceptably minimised. To confirm the minimisation of radioactive material release from the plant, severe accident progression for a range of severe accidents has been investigated and important phenomena which should be considered in severe accidents have been identified and evaluated. Specifically, the following are identified as important phenomena that could challenge the integrity of the containment boundary leading to early and large release of activity to the environment:

- Overpressure and Overtemperature of the Containment,
- Molten Core Concrete Interaction (MCCI),
- Ex-vessel Fuel Coolant Interaction (FCI), and
- Direct Containment Heating (DCH).

Against the above phenomena, PCV failure probabilities have specifically been evaluated for MCCI, ex-vessel FCI, and DCH. These evaluations, including the assessment of uncertainties associated with these phenomena, are used to support the quantification of the PCV failure modes in the Level 2 PSA.

The UK ABWR has a variety of engineered features, strategies and procedures for responding to design basis accidents and beyond design basis accidents, including severe accidents. These engineered features, together with the strategies and accident management procedures developed for the UK ABWR have also been evaluated against lessons learnt from UK, EU Stress test, and IAEA investigations on Fukushima accidents. This evaluation has provided confirmation that the UK ABWR has design provisions already included in current design or under design development to deal with these recommendations. In addition, a number of evaluations have demonstrated that the design of the UK ABWR, in relation to severe accident mitigation is consistent with Relevant Good Practice and no reasonably practicable improvements have been identified for further consideration. The ALARP assessments have been provided for the following topics:

- 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.

Furthermore, extensive analyses, including sensitivity/uncertainty analyses, have been performed for the following categories of severe accidents to confirm the robustness of the design against these accidents and the effectiveness of severe accident management measures:

- Severe accident analysis for faults at power,
- Severe accident analysis for faults at shutdown, and
- Severe accident analysis for SFP faults.

With the results and insights from these analyses, together with using the results from the UK ABWR GDA PSA analysis, an evaluation has also been performed to provide a demonstration of the ‘practical elimination’ of potential early or large fission product release for the design. Against a set of numerical criteria, the evaluation concluded that the principle of ‘practical elimination’ has largely been met by the design.

From the extensive analyses provided in the UK ABWR GDA, it can be concluded that the UK ABWR has been designed to minimise the fission product release to the environment in severe accidents and is consistent with the principle that risks are ALARP.

26.9 References

[Ref-1] Electric Power Research Institute, “MAAP4 User’s Manual”, May, 1994

[Ref-2] O.Zuchuat, et al., “Steam explosions-induced containment failure studies for SWISS nuclear power plants”, OECD/CSNI specialist meeting, JAERI, Japan, May 1997.

[Ref-3] Yuri Vasilyev, Alexander Kolodeshinivkov and Vladimir Zhdanov, “COTELS Fuel Coolant Interaction Tests under Ex-Vessel Conditions”, JAERI-Conf., 2000-2015.

[Ref-4] EC, “Review of current Severe Accident Management (SAM) approaches for Nuclear Power Plants in Europe”, JRC93082, EUR 26967 EN, 2014

[Ref-5] Electric Power Research Institute, “ADVANCED LIGHT WATER REACTOR UTILITY REQUIREMENTS DOCUMENT”, March 1999

[Ref-6] Electric Power Research Institute,”MAAP4 – Modular Accident Analysis Program for LWR Power Plants– Computer Code Manual”, December 2013

[Ref-7] Hideaki Sadamatsu, et al., “The Accident Analysis for Unit1 at Fukushima Dai-ichi Nuclear Power Station”, N9P0279, The 9th International Topical Meeting on Nuclear Thermal-Hydraulics, Operation and Safety (NUTHOS-9), 2012

[Ref-8] Shinya Mizokami, et al., “The Accident Analysis for Unit 2 at Fukushima Dai-ichi Nuclear Power Sation”, N9P0272, The 9th International Topical Meeting on Nuclear Thermal-Hydraulics, Operation and Safety (NUTHOS-9), 2012

[Ref-9] Hiromasa Yanagisawa, et al., “The Accident Analysis for Fukushima Dai-ichi Nuclear Power Station Unit 3”, N9P0326, The 9th International Topical Meeting on Nuclear Thermal-Hydraulics, Operation and Safety (NUTHOS-9), 2012

[Ref-10] Electric Power Research Institute, “Use of Modular Accident Analysis Program (MAAP) in Support of Post-Fukushima Applications”, June 2013

[Ref-11] EPRI, “Severe Accident Management Guidance Technical Basis Report Volume 1: Candidate High-Level Actions and Their Effects (Final Report)”, October 2012

[Ref-12] Office for Nuclear Regulation, "Safety Assessment Principles for Nuclear Facilities", 2014 Edition, Revision 0.

[Ref-13] IAEA, "Severe Accident Management Programmes for Nuclear Power Plants", No. NS-G-2.15, 2009

[Ref-14] "European Utility Requirements for LWR Nuclear Power Plants: Volume 2 Generic Nuclear Island Requirements: Chapter 1 Safety Requirements", Revision D, October 2012.

[Ref-15] Sandia National Laboratories, "Light Water Reactor Hydrogen Manual", NUREG/CR-2726, June 1983

[Ref-16] Office for Nuclear Regulation, "Current Reactors being Assessed", http://www.onr.org.uk/new-reactors/assessment.htm

[Ref-17] Hitachi-GE Nuclear Energy, Ltd., "Topic Report on Severe Accident Phenomena and Severe Accident Analysis", GA-91-9201-0001-00024, AE-GD-0102 Rev. H, April 2017

[Ref-18] U.S.NRC, "Design Certification Applications for New Reactors", http://www.nrc.gov/reactors/new-reactors/design-cert.html

[Ref-19] U. S. NRC, "Final Safety Evaluation Report Related to the Certification of the Economic Simplified Boiling-Water Reactor Standard Design", NUREG-1966 Volume4, April 2014.

[Ref-20] GE Hitachi Nuclear Energy, "ESBWR CERTIFICATION PROBABILISTIC RISK ASSESSMENT Chapter 21", NEDO-33201 Revision 6, October 2010.

[Ref-21] AREVA NP, "UK-EPR PCSR- PCSR – Sub-Chapter 6.2 – Containment Systems", UKEPR-0002-062 Issue 04, October 2012.

[Ref-22] GE Hitachi Nuclear Energy, "ESBWR Design Control Document Tier2 Chapter #19 PROBABILISTIC RISK ASSESSMENT AND SEVERE ACCIDENTS", 26A6642BY Revision 10, April 2014.

[Ref-23] Hitachi-GE Nuclear Energy, Ltd., "GDA Safety Case Development Manual", GA10-0511-0006-00001, XD-GD-0036 Rev. 3, June 2017.

[Ref-24] Hitachi-GE Nuclear Energy, Ltd., "GDA ALARP Methodology", GA10-0511-0004-00001, XD-GD-0037 Rev. 1, November 2015.

[Ref-25] U.S. Nuclear Regulatory Commission, "State-of-the-Art Reactor Consequence Analyses Project: Uncertainty Analysis of the Unmitigated Long-Term Station Blackout of the Peach Bottom Atomic Power Station", NUREG/CR-7155, SAND2012-10702P, May 2016.

[Ref-26] Electric Power Research Institute, "MAAP4 Applications Guidance", 1013500, October 2006

[Ref-27] A. Ohnuki, H. Kamo, and H. Akimoto, "Development of multidimensional two-fluid model code ACE-3D for evaluation of constitutive equations", JAERI-Data/Code 96-033, Japan Atomic Energy Research Institute, 1996. (In Japanese)

[Ref-28] K. Moriyama, Y. Maruyama, and H. Nakamura, "Steam Explosion Simulation Code JASMINE v.3 User's Guide", JAEA-Data/Code 2008-014, Japan Atomic Energy Agency, July 2008. http://jolissrch-inter.tokai-sc.jaea.go.jp/pdfdata/JAEA-Data-Code-2008-014.pdf

[Ref-29] ANSYS Inc. homepage, http://www.ansys.com/Products/Simulation+Technology/Structural+Analysis/Explicit+Dynamics/ANSYS+Autodyn

[Ref-30] ABE Atsushi, KATAYAMA Masahide et. al.," Numerical Simulation on Underwater Explosions and Following Bubble Pulses", Proceeding of Shock Wave Symposium (2007) pp.319-322 (In Japanese)

[Ref-31] Hiroyasu Mochizuki, "Response of a Water Filled Calandria Vessel to Explosion", 7th International Conference on Nuclear Engineering, 1999, ICONE-7018.

[Ref-32] Yoshie S, Fukuda H et. al., "Dynamic Response of a Containment Surrounding Extreme Pressure Source due to Steam Explosion", 13th International Conference on Structural Mechanics in Reactor Technology, 1995

[Ref-33] U.S. Nuclear Regulatory Commission, "Individual Plant Examination: Submittal Guidance", NUREG-1335, August 1989

[Ref-34] U.S. Nuclear Regulatory Commission, “Evaluation of Severe Accident Risks: Peach Bottom, Unit 2”, NUREG/CR-4551, SAND86-1309, Vol. 4, Rev. 1, Part 1, December 1990

[Ref-35] U.S. Nuclear Regulatory Commission, “Evaluation of Severe Accident Risks: Grand Gulf, Unit 1”, NUREG/CR-4551, SAND86-1309, Vol. 6, Rev. 1, Part 1, December 1990

[Ref-36] Theofanous TG, ‘’On the Proper Formulation of Safety Goals and Assessment of Safety Margins for Rare and High Consequence Hazards’’, Reliability Engineering and System Safety, 54, 243-257, 1996

[Ref-37] OECD/NEA/CSNI report, ‘’Recent Developments in Level 2 PSA and Severe Accident Management’’, NEA/CSNI/R (2007), November 2007

[Ref-38] R. Tokraz and R. Libby, “Recriticality in a Boiling Water Reactor Following a Core Damage Accident”, Proceedings of the 17th Water Reactor Safety Meeting, NUREG/CP-0105, Vol. 2, pp. 41-66, October 1989

[Ref-39] D. Magallon, “Characteristics of corium debris bed generated in large-scale fuel-coolant interaction experiments”, Nuclear Engineering and Design 236 (2006), 1998-2009

[Ref-40] Hitachi-GE Nuclear Energy, Ltd., “Topic Report on Fault Assessment”, GA91-9201-0001-00022 (UE-GD-0071) Rev.6, July 2017.

[Ref-41] Hitachi-GE Nuclear Energy, Ltd., “Topic Report on Beyond Design Basis Analysis”, GA91-9201-0001-00139 (AE-GD-0473) Rev.5, August 2017.

[Ref-42] Hitachi-GE Nuclear Energy, Ltd., “Topic Report on Fault Assessment for SFP and Fuel Route”, GA91-9201-0001-00082 (AE-GD-0229) Rev.3, July 2017.

[Ref-43] Office for Nuclear Regulation, “Severe Accident Safety Case (RO-ABWR-0023)”. (http://www.onr.org.uk/new-reactors/uk-abwr/reports/ro-abwr-0023.pdf)

[Ref-44] GOTHIC Thermal Hydraulic Analysis Package, Version 8.1(QA), EPRI, Palo Alto, CA: 2014.

[Ref-45] US Nuclear Regulatory Commission, Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants, 10 CFR 50 Appendix B (https://www.nrc.gov/reading-rm/doc-collections/cfr/part050/part050-appb.html).

[Ref-46] American Society of Mechanical Engineers, Nuclear Quality Assurance (NQA-1) Certification (https://www.asme.org/shop/certification-accreditation/nuclear-quality-assurance-nqa1-certification).

[Ref-47] Hitachi-GE Nuclear Energy, Ltd., "UK ABWR Nuclear Safety and Environmental Design Principles (NSEDPs)", GA10-0511-0011-00001 (XD-GD-0046) Rev.1, July 2017.

[Ref-48] Hitachi-GE Nuclear Energy, Ltd., "Topic Report on Safety Relief Valve Diversity", GA91-9201-0001-00270 (SE-GD-0601) Rev.0, April 2017.

[Ref-49] Hitachi-GE Nuclear Energy, Ltd., "Topic Report on Use of PSA in ALARP Assessment – Current Status and Future Applications", GA91-9201-0001-00232 (AE-GD-0803) Rev.2, June 2017.

[Ref-50] Hitachi-GE Nuclear Energy, Ltd., "A Study of Chemistry Effects in UK ABWR Fault Studies", GA91-9201-0003-01330 (HE-GD-0175) Rev.2, July 2017.

[Ref-51] Hitachi-GE Nuclear Energy, Ltd., "Suppression Pool pH Model during Design Basis Accident and Severe Accident", GA91-9201-0003-00451 (WPE-GD-0097) Rev.4, July 2017.

[Ref-52] Hitachi-GE Nuclear Energy, Ltd., "Basis of Safety Cases on Electrical System", GA91-9201-0002-00033 (EE-GDA-180) Rev.4, Mar 2017.

[Ref-53] Hitachi-GE Nuclear Energy, Ltd., "Basis of Safety Cases on Severe Accident Mechanical Systems", SE-GD-0219 Rev.2, GA91-9201-0002-00021, June 2017.

[Ref-54] Hitachi-GE Nuclear Energy, Ltd., "Demonstration of Practical Elimination of Early or Large Fission Product Release for UK ABWR", AE-GD-0992 Rev.0, GA91-9201-0003-02179, June 2017.

[Ref-55] Department for Business, Energy & Industrial Strategy report, The United Kingdom's Seventh National Report on Compliance with the Obligations of the Convention on Nuclear Safety, January 2017.

[Ref-56] Hitachi-GE Nuclear Energy, Ltd., "Containment Performance Analysis Report in UK ABWR", GA91-9201-0003-00985, AE-GD-0561 Rev. 3, June 2017

[Ref-57] Hitachi-GE Nuclear Energy, Ltd., "Severe Accident Safety Case for Shutdown Reactor and SFP", GA91-9201-0003-01147, AE-GD-0633, Rev.1, June 2016

Appendix A: Safety Functional Claims Tables

The relationship between the BDB and SA analysis presented in this chapter and the Safety Functional Claims (SFCs) used elsewhere in the PCSR is described below:

- This chapter of the PCSR specifies the SFCs for each of the frontline safety systems that are claimed in the fault schedule to fulfil specific high level safety functions.
- None of these SFCs are explicitly listed in this chapter. Instead they are tabulated in the Appendices of the relevant PCSR system chapters.
- BDB fault is identified from the same initiating faults as identified for DB fault and additional protection system common cause failures. BDB analysis demonstrates that there are no “CLIFF EDGE” effects and the adequacy of safety provisions based on defence in depth and diversity in the design. Therefore, SSCs for the DB faults including Class2 and Class3 SSCs can be used in the BDB faults and the similar SFCs as for DB fault measures in Table A-2 of PCSR Chapter 24 can be adopted for BDB fault. The SFCs claimed for the safety systems specifically used in BDB analysis are shown as follows.
    - SFC Tables of mechanical systems (FLSS, FLSR, RDCF, FCVS, AHEF and ANI) : Appendix A of PCSR Chapter 16
- SFCs for severe accident management measures are summarised in the following chapters.
    - SFC Tables of mechanical systems (FLSS, FLSR, RDCF, LDF, FCVS, AHEF and ANI) : Appendix A of PCSR Chapter 16
    - SFC Tables of C&I (SA C&I): Appendix A1 of PCSR Chapter 14
    - SFC Tables of electrical power supplies (BBG, Power Truck): Appendix A1 of PCSR Chapter 15 and Section 13 of BSC on Electrical System [Ref-52]
    - Human-Based Safety Claims (Severe accident management): Appendix A of PCSR Chapter 27
- SFCs for each severe accident management measures are supported by the severe accident analysis on mitigated sequences described in Section 26.4.6.2 or other assessment. The relationship between HLSF and supporting analysis is shown in Table A-1.
- The assumptions made in the beyond design basis and severe accident analyses in this chapter are consistent with the SFCs that are defined elsewhere in the PCSR as explained above.

Table A-1 Relationship between HLSF and supporting analysis

HLSF	System	Reference	Notes
2-2 Function of alternative fuel cooling	FLSS, FLSR, RDCF, BBG	PCSR Section 26.4.6.2 "Severe Accident Analysis on Mitigated Sequence"(Case 1-1, 2-1, 3-1, 4-1 of Table 26.4-4), and PCSR Section 26.4.7 "26.4.7 Severe Accident Analysis for Faults at Shutdown"	Function of core injection
2-5 Functions to make up water for spent fuel pool	FLSS, FLSR, BBG	PCSR Section 26.4.8 "Severe Accident Analysis for SFP Faults"	Function of SFP injection
3-2 Function of alternative containment cooling and decay heat removal	FCVS	PCSR Section 26.4.6.2 "Severe Accident Analysis on Mitigated Sequence" (Case 1-1, 1-2, 2-1, 2-2, 2-3, 3-1, 3-2, 4-1, 5-2 of Table 26.4-4)	Function of PCV venting
	Large Power Truck	PCSR Section 26.4.6.2 "Severe Accident Analysis on Mitigated Sequence"(Case 2-4 of Table 26.4-4)	Function of RHR recovery
	AHEF	PCSR Section 26.4.6.2 "Severe Accident Analysis on Mitigated Sequence" (Case 6-1 of Table 26.4-4)	Function of RHR recovery
4-8 Functions to minimise the release of radioactive gases	FCVS	PCSR Section 25.7.4 "Result of Level 3 PSA" (P2 and P3)	Function of fission product removal
4-9 Functions to contain radioactive materials in the event of a severe accident	FLSS, FLSR, BBG	PCSR Section 26.4.6.2 "Severe Accident Analysis on Mitigated Sequence" (Case 1-1, 1-2, 2-1, 2-2, 3-1, 3-2, 4-1, 5-2, 6-1 of Table 26.4-4)	Function of PCV spray, lower D/W injection, reactor well injection
	LDF	PCSR Section 26.4.6.2 "Severe Accident Analysis on Mitigated Sequence" (Case 2-3, 2-4 of Table 26.4-4)	Function of LDF activation
	ANI	-	Requirement of ANI is determined based on H_2/O2 generation rate by radiolysis of water.

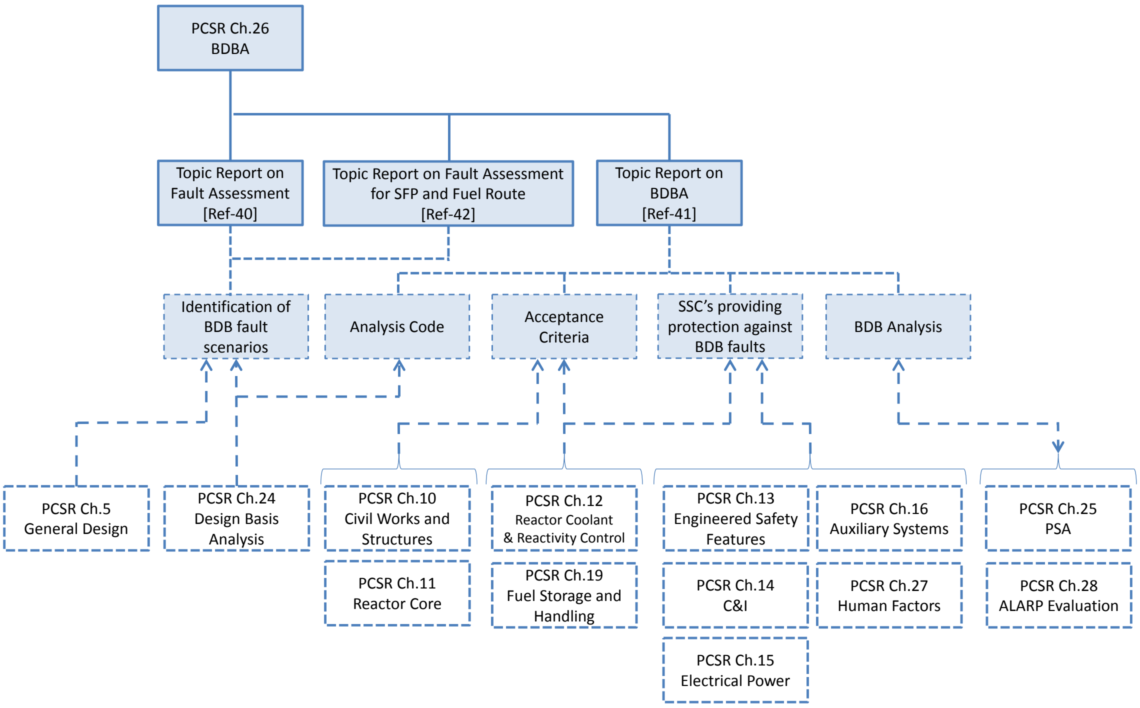
Appendix B: Safety Properties Claims Tables

The relationship between the BDB and SA analysis presented in this chapter and the Safety Properties Claims (SPCs) used elsewhere in the PCSR is described below:

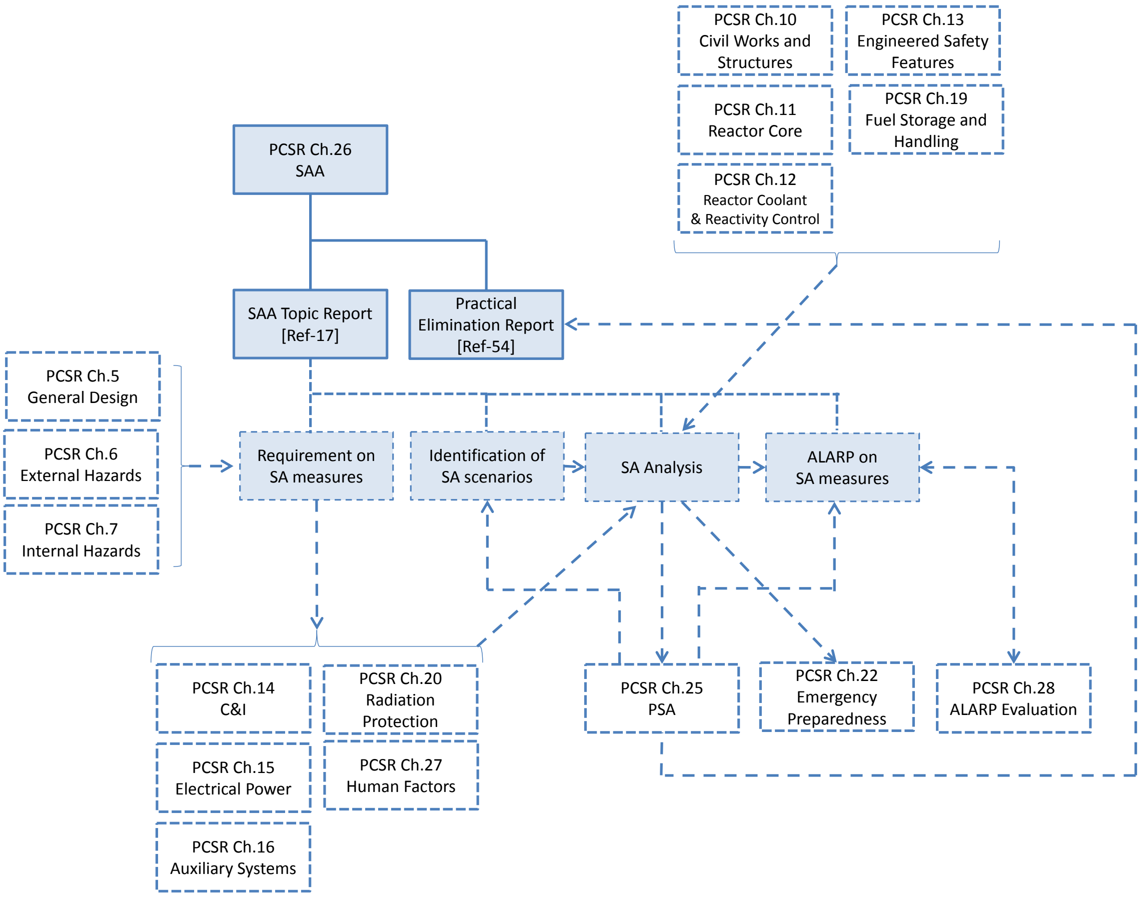
- The chapters of the PCSR that specify the SFCs also specify corresponding SPCs for each of those frontline safety systems.
- SPCs are numbered in accordance with the rules presented in the Safety Case Development Manual [Ref-23].
- None of these SPCs are explicitly listed in this chapter. Instead they are tabulated in the Appendices of the relevant PCSR system chapters.
- BDB fault is identified from the same initiating faults as identified for DB fault and additional protection system common cause failures. BDB analysis demonstrates that there are no "CLIFF EDGE" effects and the adequacy of safety provisions based on defence in depth and diversity in the design. Therefore, SSCs for the DB faults including Class2 and Class3 SSCs can be used in the BDB faults and the similar SPCs as for DB fault measures in Table B-1 of PCSR Chapter 24 can be adopted for BDB fault. The SPCs claimed for the safety systems specifically used in BDB analysis are shown as follows.
  - SPC Tables of mechanical systems (FLSS, FLSR, RDCF, FCVS, AHEF and ANI) : Appendix B of PCSR Chapter 16
- SPCs for severe accident management measures are summarised in the following chapters.
  - SPC Tables of mechanical systems (FLSS, FLSR, RDCF, LDF, FCVS, AHEF and ANI) : Appendix B of PCSR Chapter 16
  - SPC Tables of C&I (SA C&I): Appendix B of PCSR Chapter 14
  - SPC Tables of electrical power supplies (BBG, Power Truck): Appendix B of PCSR Chapter 15
  - Human Factor Property Claims (Severe accident management): Appendix B of PCSR Chapter 27
- The above SPCs are defined considering the basic requirements for severe accident management measures described in Section 26.4.3.2.
- The assumptions made in the beyond design basis and the severe accident analyses in this chapter are consistent with the SPCs that are defined elsewhere in the PCSR as explained above.

Appendix C: Document Map

Document Map for BDBA



Document Map for SAA



Appendix D: System Claims and SAA Assumptions

Tables D-1, D-2, and D-3 show the summary of the comparison between system claims and severe accident analysis assumptions for the At-power reactor, the Shutdown reactor, and the SFP. There is no inconsistency between system claims and severe accident analysis assumptions.

Table D-1 Comparison between system claims and SAA assumptions for At-power reactor (1/3)

SA Mitigation	System	SA Analysis assumptions	System claims [Ref-53]	Remarks
RPV depressurisation	RDCF	Two safety relief valves are opened at the timing when RPV water level drops below 20 percent of active fuel length from BAF. (Table 10.2-1 of [Ref-17])	The RDCF is designed with the following performance to fulfil the function to depressurise the reactor for beyond design basis scenarios without relying on the RDCF Accumulator: (1) The RDCF is capable of depressurising the RPV with two out of the four SRVs controlled by the switching valve after RCIC operation considering that their minimum discharge capacity is 460 t/h per unit. (2) The RDCF is provided with five nitrogen gas supply cylinders of 46.7 L of capacity charged at 14.7 MPa [gauge] each per train for operation of the associated switching valves. The cylinders ensure sufficient capacity to open and keep open the target SRVs under severe accident conditions when the delivery of depressurisation by RDCF through the accumulator is not possible for seven days (maximum pressure in the D/W is assumed to be 2Pd).	SA analysis assumption is supported by the system claim.
Alternative core injection	FLSS	Core injection is assumed to be conducted at 60 minutes after the accident with the flowrate of 90 m^3/hr. (Table 10.2-1 of [Ref-17])	The FLSS is capable of injecting a coolant flowrate of 90 m^3/h into the RPV against a RPV pressure of 0.62 MPa[gauge].	SA analysis assumption is supported by the system claim.
	FLSR		The FLSR is capable of injecting a coolant flowrate of 90 m^3/h into the RPV against a RPV pressure of 0.62 MPa[gauge] (same requirement as the FLSS).	SA analysis assumption is supported by the system claim. In PSA, FLSR is taken credit only in scenarios where core injection is conducted after 8 hours from the start of an accident (e.g. TB sequence).
Lower D/W injection (before RPV failure)	FLSS	Lower drywell injection is conducted when the lower head temperature reaches 300 ^oC. Flow rate is 90 m^3/hr and water level of 2 m is formed in the lower D/W. (Table 10.2-1 of [Ref-17])	The FLSS is capable of injecting a coolant flowrate of 90 m^3/h into the lower D/W against a lower D/W pressure of 0.62 MPa[gauge] for making a pool before RPV failure.	SA analysis assumption is supported by the system claim.
	FLSR		The FLSR is capable of injecting a coolant flowrate of 90 m^3/h into the lower D/W against a lower D/W pressure of 0.62 MPa[gauge] for making a pool before RPV failure.	SA analysis assumption is supported by the system claim. In PSA, FLSR is taken credit only in scenarios where lower D/W injection is conducted after 8 hours from the start of an accident.

Table D-1 Comparison between system claims and SAA assumptions for At-power reactor (2/3)

SA Mitigation	System	SA Analysis assumptions	System claims [Ref-53]	Remarks
Lower D/W injection (after RPV failure)	FLSS	After RPV failure occurs, the flow rate is controlled to be amount of water required to remove the decay heat. Specifically, flow rate is assumed as follows. 0 - 5 hr : 70 m^3/hr 5 - 10 hr : 50 m^3/hr 10 - 20 hr : 45 m^3/hr ... (Table 10.2-1 of [Ref-17])	The FLSS is capable of injecting a coolant flowrate of 40 m^3/h into the lower D/W against a lower D/W pressure of 0.62 MPa[gauge] for molten core cooling after RPV failure. Total coolant flowrate of 70 m^3/h is required to cool the molten core, thus a coolant flowrate of 30 m^3/h is provided with RPV injection.	SA analysis assumption is supported by the system claim.
	FLSR		The FLSR is capable of injecting a coolant flowrate of 40 m^3/h into the lower D/W against a lower D/W pressure of 0.62 MPa[gauge] for molten core cooling after RPV failure. Total coolant flowrate of 70 m^3/h is required to cool the molten core, thus a coolant flowrate of 30 m^3/h is provided with RPV injection.	SA analysis assumption is supported by the system claim. In PSA, FLSR is taken credit only in scenarios where lower D/W injection is conducted after 8 hours from the start of an accident.
	LDF	The flow path of the ten valves of the LDF system is opened passively when the gas temperature of the lower D/W reaches 260 °C. (Table 10.2-1 of [Ref-17])	The LDF is designed with the following performance to fulfill the function to cool the molten core: (1) The LDF is capable of providing a coolant flowrate of 22 L/s. (which is achieved by two of ten LDF injection lines) (2) The LDF is initiated automatically with the LDF Fusible Plugs when lower D/W temperature reaches around 260 °C.	The number of valves to open is inconsistent, but the system claim of the LDF system is consistent ‘practically’ because there is no impact on the effectiveness evaluation for the LDF system even if the number of valves to open in the LDF system is changed from ten to two.
Upper D/W spray	FLSS	PCV spray is conducted at 60 minutes after the accident with the flowrate of 300 m^3/hr. (Table 10.2-1 of [Ref-17])	The FLSS is capable of injecting a coolant flowrate of 300 m^3/h into the PCV against a PCV pressure of 0.62 MPa[gauge] (same requirement as the FLSR).	SA analysis assumption is supported by the system claim.
	FLSR		The FLSR is capable of injecting a coolant flowrate of 300 m^3/h into the PCV against a PCV pressure of 0.62 MPa[gauge] (same requirement as the FLSS).	SA analysis assumption is supported by the system claim. In PSA, FLSR is taken credit only in scenarios where PCV spray is conducted after 8 hours from the start of an accident.

Table D-1 Comparison between system claims and SAA assumptions for At-power reactor (3/3)

SA Mitigation	System	SA Analysis assumptions	System claims [Ref-53]	Remarks
Reactor well injection	FLSS	Reactor well injection is assumed to be conducted at 8 hours after the accident. Flow rate is 70 m^3/hr and water level of 1.6 m is formed in the reactor well. (Appendix C of [Ref-56])	The FLSS is capable of injecting a coolant flowrate of 70 m^3/h into the reactor well against a reactor well pressure of 3.43 kPa[gauge]. (same requirement as the FLSS)	SA analysis assumption is supported by the system claim.
	FLSR		The FLSR is capable of injecting a coolant flowrate of 70 m^3/h into the reactor well against a reactor well pressure of 3.43 kPa[gauge].	SA analysis assumption is supported by the system claim.
Decay heat removal	AHEF	RHR is restored by AHEF. Heat removal rate is 23 MW. (Table 10.2-1 of [Ref-17])	Heat removal capacity: 23 MW (/division)	SA analysis assumption is supported by the system claim.
	FCVS	W/W venting is conducted at the timing when PCV pressure reaches 620 kPa[gauge]. Mass flow rate of steam corresponding to 1 percent decay heat at 310 kPa[gauge] is assumed. (Table 10.2-1 of [Ref-17])	The FCVS is capable of venting a PCV gas flowrate of 15.8 kg/s at PCV pressure of 0.31 MPa[gauge]. The rupture disc of the COPS bursts when PCV pressure reaches twice of design value (2Pd).	SA analysis assumption is supported by the system claim. The detail setting pressure of the COPs will be determined after GDA.
PCV capacity	V/B	No failure except for high RPV failure scenario (Appendixes E and F of [Ref-56])	The ultimate capacity of the containment is summarised in PCSR Section 10.6.4. Vacuum breakers are unlikely to fail in severe accident.	SA analysis assumption is supported by the supporting document.
	RCCV	Overpressure and overtemperature: 620 kPa[gauge], about 300 °C failure area : 0.068 m^2 (Section 3.4 of [Ref-56])	The ultimate capacity of the containment is summarised in PCSR Section 10.6.4. It bounds the PCV failure criteria given by Figure 26.4-4.	SA analysis assumption is supported by the supporting document.

Table D-2 Comparison between system claims and SAA assumptions for Shutdown reactor

SA Mitigation	System	SA Analysis assumptions	System claims [Ref-53]	Remarks
RPV depressurisation	RDCF	Two safety relief valves are opened at the timing when RPV water level drops below 20 percent of active fuel length from BAF. (Table 4.6-2 of [Ref-57])	The RDCF is designed with the following performance to fulfil the function to depressurise the reactor for beyond design basis scenarios without relying on the RDCF Accumulator: (1) The RDCF is capable of depressurising the RPV with two out of the four SRVs controlled by the switching valve after RCIC operation considering that their minimum discharge capacity is 460t/h per unit. (2) The RDCF is provided with five nitrogen gas supply cylinders of 46.7L of capacity charged at 14.7MPa [gauge] each per train for operation of the associated switching valves. The cylinders ensure sufficient capacity to open and keep open the target SRVs under severe accident conditions when the delivery of depressurisation by RDCF through the accumulator is not possible for seven days (maximum pressure in the D/W is assumed to be 2Pd).	SA analysis assumption is supported by the system claim.
Alternative core injection	FLSS	Alternative core injection is assumed to start after the core support plate failure. Water flow rate is assumed to be 90 m^3/hr. (Table 4.6-2 of [Ref-57])	The FLSS is capable of injecting a coolant flowrate of 90 m^3/h into the RPV against a RPV pressure of 0.62 MPa[gauge].	SA analysis assumption is supported by the system claim.
	FLSR		The FLSR is capable of injecting a coolant flowrate of 90 m^3/h into the RPV against a RPV pressure of 0.62MPa[gauge] (same requirement as the FLSS).	SA analysis assumption is supported by the system claim. In PSA, FLSR is taken credit only in scenarios where lower core injection is conducted after 8 hours from the start of an accident or FLSR deployment is finished in advance.
Lower D/W injection (before RPV failure)	FLSS	–	–	These systems can be used in a severe accident during shutdown. However, as these systems are not taken credit in PSA, no analysis is conducted.
	FLSR	–	–	
Lower D/W injection (after RPV failure)	FLSS	–	–	
	FLSR	–	–	
	LDF	The flow path of the ten valves of the LDF system is opened passively when the gas temperature of the lower D/W reaches 260 °C. (Table 4.6-2 of [Ref-57])	The LDF is designed with the following performance to fulfill the function to cool the molten core: (1) The LDF is capable of providing a coolant flowrate of 22 L/s. (which is achieved by two of ten LDF injection lines) (2) The LDF is initiated automatically with the LDF Fusible Plugs when lower D/W temperature reaches around 260 °C.	The number of valves to open is inconsistent, but the system claim of the LDF system is consistent ‘practically’ because there is no impact on the effectiveness evaluation for the LDF system even if the number of valves to open in the LDF system is changed from ten to two.
Upper D/W spray	FLSS	–	–	These systems can be used in a severe accident during shutdown. However, as these systems are not taken credit in PSA, no analysis is conducted.
	FLSR	–	–	
Reactor well injection	FLSS	–	–	
	FLSR	–	–	
Decay heat removal	AHEF	–	–	
	FCVS	–	–	If PCV boundary can be re-isolated, these systems can be used. However, as re-isolation of PCV boundary is not taken credit in PSA, no analysis is conducted.
PCV capacity	V/B	–	–	
	RCCV	–	–	

Table D-3 Comparison between system claims and SAA assumptions for the SFP

SA Mitigation	System	SA Analysis assumptions [Ref-17]	System claims [Ref-53]	Remarks
SFP spray	FLSS	In catastrophic failure scenario, SFP spray is assumed to be conducted from 0.5 hours after an accident with the flowrate of 120 m^3/hr. In other scenarios, SFP spray is assumed to be conducted from 0.5 hours after fuel damage. (Table 3.6-2 of [Ref-57])	The FLSS is capable of injecting a coolant flowrate of 120 m^3/h into the SFP against a SFP pressure of 3.43kPa[gauge].	SA analysis assumption is supported by the system claim.
	FLSR		The FLSR is capable of injecting a coolant flowrate of 120 m^3/h into the SFP against a SFP pressure of 3.43 kPa[gauge].	SA analysis assumption is supported by the system claim. In PSA, FLSR is taken credit only in scenarios where SFP spray is conducted after 8 hours from the start of an accident.