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UK ABWR

Document ID	:	GA91-9101-0101-01000
Document Number		XE-GD-0214
Revision Number	:	C

UK ABWR Generic Design Assessment

Generic PCSR Chapter 1 : Introduction

Hitachi-GE Nuclear Energy, Ltd.

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Executive Summary

The Generic Pre-Construction Safety Report (Generic PCSR) is a central document within the process of the Generic Design Assessment (GDA), through which Hitachi-GE Nuclear Energy, Ltd. (Hitachi-GE) outlines the reasons why it believes that the UK Advanced Boiling Water Reactor (UK ABWR) can be safely operated in the UK. Specifically this PCSR provides substantiation of the following overarching safety case claim:

A UK ABWR constructed on a generic site within the United Kingdom (UK), meets all safety targets for the public, workers and the environment, and satisfies the principle that all risks are as low as reasonably practicable (ALARP) for all operating and fault conditions.

The submission of the UK ABWR Generic PCSR to the UK Regulators marks a significant milestone in the GDA process.

This introductory chapter does not contain technical summaries of the reactor design itself. These can be found on the UK ABWR website (www.hitachi-hgne-uk-abwr.com), which was launched at the start of GDA Step 2, and in Generic PCSR Chapter 9. Rather, this chapter gives an overview of the contents of the Generic PCSR and explains Hitachi-GE's approach to meeting regulatory expectations for GDA. It outlines the hierarchy of the Generic PCSR documentation; breaking down, to some extent, the types of information it contains. It is, in effect, a readers' guide to the Generic PCSR and an indicative overview of the type of information it contains. It also includes a list of abbreviations and acronyms that are often referred to in the other Generic PCSR chapters as well as in lower tier PCSR documentation.

By its very nature, Chapter 1 is technical and is based on regulatory terminology, such as, the Claims, Arguments, Evidence approach that has been adopted in the safety case for many topic areas. There is also an element of assumed-knowledge, based on information contained in Steps 1, 2, 3 of UK ABWR GDA submissions which Hitachi-GE has published for the UK ABWR. However, when read in conjunction with the appendices, this chapter provides an overview of the process, structure and approach for development of Hitachi-GE's Generic PCSR and related GDA submissions as a whole.

Chapter 1 also provides a route map of the Generic PCSR by providing brief summaries of the purpose and roles of each of the PCSR chapters.

1.1 Introduction

The Generic Pre-Construction Safety Report (Generic PCSR) is a key document and submission within Generic Design Assessment (GDA) that outlines the reasons supporting the top level claim that Hitachi-GE believes *"the UK Advanced Boiling Water Reactor (UK ABWR) constructed on a generic site within the United Kingdom (UK), can be operated safely under all operating and fault conditions."*

The Generic PCSR has been submitted at the end of GDA Step 2, 3. The final version of PCSR Rev. C planned in Step 4 marks a significant milestone in GDA and paves the way for ONR's decision on whether or not to grant a Design Acceptance Confirmation (DAC).

Generic PCSR Chapter 1, Introduction, provides an overview of the Generic PCSR, explains the approach for and hierarchy of UK ABWR safety cases, provides a route map to the PCSR by providing summaries of the contents of each of the Generic PCSR chapters.

1.1.1 Background

The top level claim is underpinned by the following high level claims:

- The UK ABWR design is based on sound international practices and over 40 years of Hitachi-GE experience in design, construction and maintenance of Boiling Water Reactors [Ref-5], [Ref-7] and [Ref-18] section 1.6.
- All credible faults and hazards have been identified and assessed.
- The safety functions of the UK ABWR are clearly identified and understood.
- The design of the UK ABWR includes suitable and sufficient Structures, Systems and Components (SSCs) to deliver the essential safety functions.
- The Hitachi-GE GDA Safety Management Arrangements are adequate.
- The risks associated with the design, construction, commissioning, operation, decommissioning of the UK ABWR are ALARP.

The first of these high-level claims is supported by Chapter 28 of this PCSR, which traces the development of the UK ABWR from the earliest Boiling Water Reactors (BWRs) and the programme of Advanced Boiling Water Reactors (ABWRs) in Japan and shows that, at each stage of development, the most appropriate technology available has been deployed; the design has been informed by the most up-to-date operational experience; and that each development has progressively reduced potential risks.

The second of these high level claims (credible faults and hazards have been identified and assessed) is supported by the work on design basis accidents, probabilistic safety analysis and Beyond Design Basis and Severe Accident Analysis (Chapters 24 to 26), and the chapters on External and Internal Hazards (Chapters 6 and 7 respectively). The hazards are grouped in to bounding faults that are used to link the hazards chapters, analysis chapters, and systems chapters as described in section 1.4.2. These chapters demonstrate the tolerance of the UK ABWR to faults and hazards, and conclude that all relevant risk targets are met. Radiological safety during normal operations is discussed in Chapter 20, and confirmed to meet all relevant targets.

The third high level claim is that Safety Functions are clearly identified and understood. The UK ABWR safety case uses a rigorous system of safety claims (Safety Functional Claims (SFCs), Safety Properties Claims (SPCs) and Human Based Safety Claims (HBSCs)) which clearly define the safety requirements. This enables a clear link to be made between safety requirements and the SSCs that deliver them. The SSC requirements are primarily defined and identified within the 'Systems' chapters in the PCSR (principally Chapters 10 to 19).

The fourth high level claim is that the UK ABWR includes suitable and sufficient SSCs to deliver the required safety functions. This claim is supported by a range of 'Systems' chapters within the PCSR covering all SSCs that are important to safety. These chapters (principally Chapters 10 to 19) provide descriptions of plant structures, systems and components, list the associated safety claims and provide links to where more detailed safety justification can be found for each SSC. The delivery of the functions is supported by aspects of human machine interface, and human factors are provided through Chapters 21 and 27, respectively.

The fifth high level claim is supported by Chapter 4, which describes how Hitachi-GE's generic safety management arrangements contribute to ensuring that the required levels of nuclear safety will be delivered throughout the lifetime of the UK ABWR (design, construction, commissioning, operation, and decommissioning). It also describes how these safety management arrangements will support the future operator and licensee in meeting safety and environment limits and conditions, including describing the arrangements for sharing knowledge of the UK ABWR technology.

The final high level claim that risks have been reduced ALARP is underpinned by Chapter 28. The overall ALARP case in Chapter 28 draws on evidence from across the wider safety case. To improve the clarity of this process, all relevant chapters of the PCSR include a section titled 'Summary of ALARP Justification' which aims to capture specific examples of how the ALARP principle has been applied for that topic area.

1.1.2 Document Structure

Chapter 1 in itself will not provide specific supporting information and/or justifications of these arguments, but will provide high level pointers to where these arguments are supported and justified by the information provided in the Generic PCSR Chapters 2 to 32, and the supporting information in each of these chapters. The overall structure of the Generic PCSR, which provides these high level pointers as explained in section 1.4.

This chapter includes the following sections:

Section 1.1 Introduction: This section introduces the PCSR generally and Chapter 1 specifically. It also presents an overview of the high level claims for the UK ABWR, including a top level claim, high level sub-claims and links to where supporting arguments can be found within the PCSR.

Section 1.2 Purpose and Scope of Chapter 1: This section sets out the purpose of the chapter and identifies what is included in the scope of the chapter and what is excluded.

Section 1.3 Purpose of the Generic Design Assessment (GDA): This section describes the GDA process with reference to corresponding regulatory guidance. This section also describes the UK ABWR's achievements through the GDA process.

Section 1.4 Structure and Contents of the UK ABWR Generic PCSR: This section describes the structure of the Generic PCSR. It explains the hierarchy of documentation that make up the wider safety case and outlines what kind of information is provided in each level of documentation (e.g. top level PCSR and supporting lower tier documents). This section also describes Hitachi-GE's approach to Claim-Argument-Evidence for GDA and describes the key elements developed to assist production of a clear, consistent, traceable and manageable safety case (e.g. Safety Case Development Manual (SCDM), Nuclear Safety and Environmental Design Principles (NSEDPs)). This section also describes the scope of the Generic PCSR.

Section 1.5 PCSR User Guide and Route Map: This section describes what kind of information can be found in each chapter of the PCSR. It provides a link or 'route map' to where information can be found within the PCSR. This section also describes the generic structure within each chapter. This section also refers to a technical summary of the contents of each chapter, which is included in each of the chapters as Executive Summary.

Section 1.6 Safety Assessments and Achievements of the ABWR Design Prior to the UK ABWR: This section is a brief summary of the licensing and construction achievements of the ABWR plant design prior to the UK ABWR.

Section 1.7 Abbreviations and Acronyms List: This section refers to the list of abbreviations and acronyms that will often be referred to in the Generic PCSR (this list is included as Appendix A).

1.2 Purpose and Scope of Chapter 1

1.2.1 Purpose

This document provides an introduction to the Generic PCSR for the UK ABWR. Its objectives are to:

- Describe the overall purpose and scope of the PCSR.
- Provide summary of the GDA assessment process and the key achievements accomplished by the UK ABWR.
- Present an overview of the high level safety claims for UK ABWR, by setting out the top level claim and the underlying sub claims that a UK Advanced Boiling Water Reactor (UK ABWR) can be operated safely in the UK.
- Describe the GDA process with reference to corresponding regulator guidance, and provide a summary of the assessment results of previous GDA Steps. Provide a high level summary and link to the specific chapters of the PCSR that will demonstrate that the sub claims are substantiated. This provides a route map for the safety case.
- Describe the high level principles, objectives and strategic approach used by Hitachi-GE to develop the Generic PCSR. This includes describing how a robust and traceable Claim-Argument-Evidence approach has been applied to the PCSR.
- Describe the hierarchical structure of GDA documentation (e.g. Level 1, Level2 and Level 3) and describe the main building blocks of the safety case (e.g. Topic Reports, Basis of Safety Case).
- Provide a description of the process undertaken to define the GDA scope.
- Provide a summary of licensing and construction achievements of the ABWR plant design prior to the UK ABWR to support the overall safety justification for UK ABWR.
- Provide a list of abbreviations and acronyms that are often be referred to in the Generic PCSR documentation as well as lower tier documentation.

1.2.2 Scope

Chapter 1 is an introduction to the Generic PCSR, and provides an overview of what can be found in the other chapters of the PCSR. Chapter 1 does not include specific supporting information and/ or justification of the safety arguments for the UK ABWR, but provides pointers to where supporting arguments are developed and further justification can be found in Chapters 2 to 32.
The overall scope of the Generic PCSR is described in Section 1.4.4

1.3 Purpose of the Generic Design Assessment (GDA)

Generic Design Assessment (GDA) is a voluntary step toward obtaining consent to build a nuclear power plant in the UK. The GDA process has been adopted by the Office for Nuclear Regulation (ONR) and the Environment Agency, and is described in detail in [Ref-1] and [Ref-2].

As described in detail in, "New nuclear reactors: Generic Design Assessment Guidance to Requesting Parties" (GDA Guidance) [Ref-1], the objective of the GDA is to allow the Regulators to undertake an assessment of the design significantly in advance of planned construction. This allows them to identify any possible shortfalls with regard to the safety, security and environmental requirements in the UK in relation to the design, which would require design changes and modifications. This process not only contributes to the safety of the plant but also reduces the risk of project delays at a later date.

Upon successful completion of the GDA assessments, the Requesting Party (RP) is issued with a Design Acceptance Confirmation (DAC) (which is valid for 10 years from the date of issue) from ONR, and a Statement of Design Acceptability (SoDA) from the Environment Agency.

GDA is a four step process, and each constitutes different aspects of assessment. These are shown in summary below, while the detailed aspects and processes are shown fully in the GDA Guidance [Ref-1].

Step 1: Preparation and submission of preliminary safety case

This is the initial engagement between the ONR, Environment Agency and the RP, Hitachi-GE, to initiate the GDA process. During this step, the RP is expected to develop an understanding of the technical and project management processes and requirements of the GDA. During this step, the UK Regulators make a decision on whether the RP is ready to proceed to the second step of the GDA process. The GDA Step 1 assessments for the UK ABWR were carried out between April 2013 and December 2013.

Step 2: Fundamental design, safety and security case claims overview

Step 2 involves assessment of the fundamental safety cases, security and environmental claims made by the RP, to confirm the soundness of the design and to identify any significant shortfalls in the design. In this step, the UK Regulators initiate a comprehensive assessment of the proposed design. Hitachi-GE submitted the UK ABWR Preliminary Safety Reports (PSRs) to ensure that the Regulators had sufficient information to carry out the assessments.

Additionally, during the course of Step 2, Hitachi-GE responded to inquiries, such as Regulatory Queries (RQs) and Regulatory Observations (ROs), issued by the Regulators.

At the end of Step 2, Hitachi-GE produced the initial Generic PCSR which comprised a summary of claims in the corresponding Level 2 and 3 documentation, and brief justification of the claims. The Step 2 Generic PCSR also took into account comprehensive interactions with the Regulators during the entire Step 2. A key interaction was in the development and agreement of the structure of the

Generic PCSR, where the RP and the regulators agreed on a structure that was not only sufficient for Step 2 but also that could readily scale well to accommodate Step 3 and 4 assessments.

At this stage the Regulators also confirmed that they had not identified any reason why the design's compliance with the legal duty in Great Britain to ensure that risks to workers and the public arising from the operation of a power station are reduced to 'So Far As Is Reasonably Practicable' (SFAIRP) or ALARP could not be demonstrated.

The second step also involved the assessment of relevant quality assurance arrangements to support the development of safety cases and claims documentation for the related design. Specifically, the regulatory audit and the readiness review of the relevant project arrangements and quality assurance arrangements concluded that Hitachi-GE had in place the necessary arrangements to cover Step 2 and the arrangements were applicable or could be readily scaled into Step 3 and beyond.

Additionally, considering the importance of transparency in the GDA process, the RP is required to involve the UK public in the GDA process through publication of relevant assessment documentation. To this end, the RP is required to run a website-based comment process, through which it receives and responds to public comments and questions; as well as publishing extensive GDA documentation (taking into account security and commercial sensitivities). This has been underway since the beginning of Step 2, and will continue through Step 4.

Based on the individual activities, described above that were carried out in the course of Step 2, and the corresponding regulatory conclusions above, the Regulators provided summaries of Step 2 Assessment of the UKABWR [Ref-12] [Ref-14], in which:

- ONR considered the fundamental safety and security aspects of the design, and the Environment Agency considered the environmental acceptability of the design and concluded that the UK ABWR assessment could move through the subsequent steps of the GDA process.
- And more importantly, at the end of Step 2 assessments the Regulators confirmed that the assessments did not identify any fundamental safety or security issues that might prevent issue of a DAC or that would need to be addressed in order to acquire one [Ref-12] and have not at this stage identified any matters addressed by the submission that are obviously unacceptable [Ref-14].

Step 3: Overall design, safety and security case arguments review

Step 3 involves detailed assessment of the overall design at the systems level. This includes analysis of safety cases, and the security and environment related arguments of the corresponding design, as well as assessment of Arguments related to ALARP.

The RP is also required to define the detailed technical scope of the GDA application, define a Design Reference and a Design Reference Point (DRP), and define the corresponding supporting documentation (included in the Master Document Submission List - MDSL) which would eventually be included in the DAC and/or SoDA.

To this effect, at the end of Step 2, the RP provided the regulators with initial information on Design Reference [Ref-9] and the initial Master Document Submission List (MDSL) [Ref-10]. The Design Reference is based on a Japanese ABWR design (such as [Ref-5]) which is the base for the UK ABWR.The MDSL was compiled based on the supporting documents recorded in the Step 2 Generic PCSR and its supporting documentation. The MDSL will be regularly updated to reflect and capture the latest design information.

Additionally, toward defining the Design Reference Point at the end of Step 3, the RP produced a document explaining the process it intends to follow to achieve this in the early stage of Step 3 [Ref-9].

In the course of Step 3, the regulators undertook a detailed review of the UK ABWR design, and produced RQs, ROs where additional information was necessary or where the regulators considered could be areas of potential shortfalls and required deep scrutiny at an early phase in the GDA process.

The discussions between the RP and the regulators in the course of Step 3 led to the production or revision of Level 2 documentation such as Topic Reports and Basis of Safety Cases (see section 1.3) that would underpin the claims of the UK ABWR. These along with other supporting documents have been submitted to the Regulators to assist them to carry out the necessary detailed Step 3 assessments.

Step 4: Detailed design, safety and security case assessment

Step 4 involves in-depth assessment of corresponding Evidence to demonstrate the safety of the design, as well as corresponding security and environmental arrangements. The RP is required to provide corresponding evaluation and analysis results as evidence to demonstrate that the design meets the Claims and Arguments described in the preceding Steps 2 and 3.

In the process of Step 4 planning, the RP intends to develop a programme of meetings, documentary scope etc covering the requirements described in the Guidance to Requesting Parties [Ref-1] as well as the specific technical areas that the Regulators intend to assess as recorded in their Step 4 assessment plans.

Upon successful completion of Step 4, which marks the completion of the GDA process, the RP will be issued a DAC and a SoDA from the ONR and the Environment Agency, respectively.

1.4 Structure and Contents of the UK ABWR Generic PCSR

This section provides a summary of the key aspects in the development of GDA documentation. One aspect is a summary of how the Generic documentation is structured. Another aspect is the Claim-Argument-Evidence (CAE) approach adopted in UK ABWR safety cases to ensure that interface management between the large number of documents in the different levels of the GDA documentation structure are defined (established) and linked together to achieve the construction of a consistent, coherent, and manageable safety case. Another aspect is the structure of the Generic PCSR and its contents. A further aspect explained in this section is the approach adopted to define and develop the scope of GDA. Further to the scope of GDA, this section also provides a summary on how the initial phases of the specific site scope are developed in the Generic PCSR with the aim of de-risking the site specific licensing phase.

1.4.1 PCSR Documentation

The GDA documentation is structured in levels to clearly present the overall safety cases in a consistent and coherent manner.

The overall documentation structure is divided into 3 levels as shown in Figure 1.4-1.

- Level 1 : SSE submissions

Level 1 forms the top-tier documentation and consists of the SSE submissions: The Generic PCSR, the Generic Environmental Permit (GEP) documentation and the Conceptual Security Arrangements (CSA). In the Generic PCSR, safety, security and environmental Claims of the UK ABWR design are illustrated. The high level safety Claims are developed in a way which demonstrates at a high level that the UK ABWR design meets UK safety, security and environmental requirements, and that the risks associated with the design are ALARP. The descriptions here after concentrate on the structure of the safety cases – the PCSR.

As part of creating a consistent, coherent and easy to maintain safety case, as required in ONR Nuclear Safety Technical Assessment Guide (TAG) 51 [Ref-16], the Claims in the Generic PCSR are clearly defined and linked/referenced to the corresponding supporting information in Level 2 and 3 documentation.

Hitachi-GE's Safety Case documentation consists of three types of claims : Safety Functional Claims (SFCs), Safety Properties Claims (SPCs) and Human Based Safety Claims (HBSCs). The approach adopted in development of safety cases using these claims is described in Section 1.4.2.

The structure of the Generic PCSR is described in the succeeding Section 1.4.3 Structure and Contents. On the other hand, the detailed structure and contents of the GEP documentation is shown in "Summary of the Generic Environmental Permit Applications" GA91-9901-0019-00001 (XE-GD-0094) Rev. G. The structure of Security documentation is in line with Guidance on the Security Assessment of Generic New Nuclear Reactor Designs [Ref-19] and corresponding arrangemetns between the Regulators and Hitachi-GE.

- Level 2 : Supporting Documentation

Level 2 consists of the Arguments documentation used to support and substantiate the Claims in Level 1. Level 2 is the first level of supporting documentation and provides the linkage between Claims in Level 1 and Evidence in Level 3 documentation. These documents comprise of Topic Reports (TRs) on typical GDA assessment areas identified in the GDA Guidance [Ref-1], and Basis of Safety Cases (BSCs) reports on the Systems, Structures and Components (SSCs) of the UK ABWR design. The list of TRs and BSCs to support UK ABWR claims were determined during Step 2 assessments, and were then developed, submitted and / revised during Step 3 and 4. The present list as well as the revision status of the TRs and BSCs is shown in the latest version of the MDSL [Ref-10]. TRs and BSCs are living documents that were developed or revised further to capture technical discussions in the course of Step 4. The contents of TRs and BSCs have been consolidated with reference to the supporting information included in the MDSL in June 2017. The contents of the PCSR and BSCs/TRs have also be consolidated before the submission of PCSR Rev. C.

The TRs, and BSCs are used to capture, in totality, the key aspects claimed on the corresponding SSCs using clearly defined and uniquely numbered Claims, and create an audit trail that links or will be used link the justifying/supporting information in the lower tier documents.

Additionaly, the BSCs, and TRs also provide comprehensive information, referencing and/or pointers to detailed information on assumptions, operating limits and conditions that will not only be used in GDA but on a wider aspect as the building blocks for arrangements for moving the GDA to the operating regime in accordance with procedure for identification of Assumptions, Limits and Conditions for Operation [Ref-17].

A procedure on the standard structure and contents of the TRs, and BSCs has already been developed and used in development of BSCs and TRs [Ref-13].

- Level 3: Supporting Documentation

Level 3 includes detailed design, evaluation and analysis documentation that will be used to provide Evidence to support and substantiate the Arguments in Level 2 and demonstrate that the UK ABWR design meets the claims in Level 1. Additionally, Level 3 documentation may not be referenced in Level 1, it may include documents generated by external organisations, and in some cases such documents may be generated outside the scope of the UK ABWR GDA.

- Others: Safety Case, Project Arrangements and Procedures, and Management of Safety and Quality Assurance (MSQA) Documentation.

GDA documentation also consists of GDA Safety Case development guidelines, project arrangements and/or procedures, and Management System and Quality Assurance (MSQA) Documents created and submitted for use in the GDA project. These documents are used to support GDA submissions where referenced accordingly.

The key safety case development guidelines include:

- Safety Case Development Plan [Ref-20]

  The SCDP describes Hitachi-GE's approach to developing the Generic PCSR, and how the PCSR will change over time.

- Safety Case Development Manual (SCDM) [Ref-18]

  The SCDM provides Hitachi-GE's GDA safety case authors, reviewers and approvers with guidance on how to achieve a sound, well-presented safety case, which is clear, coherent and consistent.

- Nuclear Safety and Environmental Design Principles (NSEDPs) [Ref-22]

  These are the high level safety and environmental principles that apply to UK ABWR, and that the design will be judged against.

- GDA ALARP Methodology [Ref-21]

  This document sets out Hitachi-GE's approach to ensuring that risks have been reduced to ALARP. It provides more detailed, practical guidance than the SCDM, including a range of suggested approaches from simple qualitative methods to more detailed ranking and rating exercises.

The relevant MSQA arrangements supporting the production of safety cases are described within or through Generic PCSR Chapter 4.

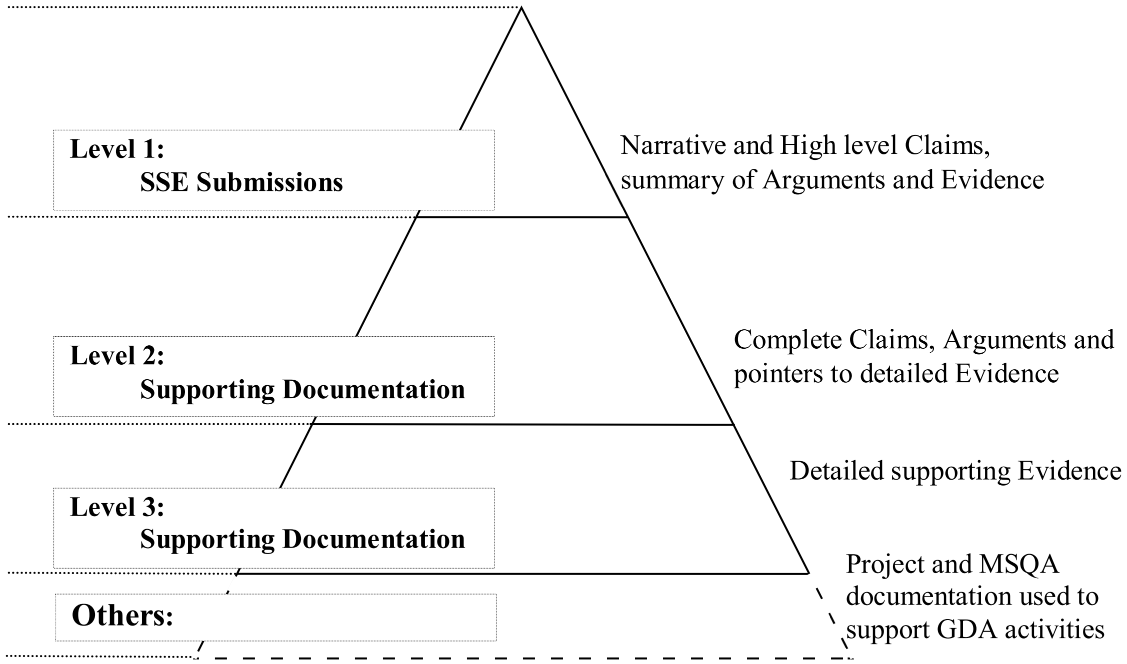

Figure 1.4-1: Hierarchy of Generic PCSR Documentation

1.4.2 Approach to Claim-Argument-Evidence (CAE)

The Claim-Argument-Evidence (CAE) approach is commonly used in structuring safety cases in the nuclear industry and elsewhere. CAE is not mandatory for GDA or in the UK generally, but the ONR Guidance to Requesting Parties [Ref-1] supports its use.

CAE is a structured approach which aims to improve the clarity of the safety case, and make complex safety justifications more understandable. The terms Claim-Argument-Evidence can be defined as:

Claim:	High level proposition, assertion or statement
Argument:	The reason why the claim is justified Supports the claim and points to where the supporting evidence can be found
Evidence:	Facts and judgements that support the argument

Hitachi-GE has adopted the CAE approach (where relevant as defined in SCDM [Ref-18] Section 3) for the UK ABWR safety case, and have defined three main types of claims: Safety Functional Claims (SFCs) and Safety Properties Claims (SPCs) and Human Based Safety Claims (HBSCs):

- Safety Functional Claims (SFCs):

  SFCs are actions performed by an SSC to implement a safety function (e.g. insert control rods, open a valve, start a pump). SFCs define the functional claims on SSCs that must be justified in order to demonstrate that the plant is safe.
  SFCs are derived from the detailed and wide ranging fault and hazard analysis carried out to support the UK ABWR, and are justified within the Basis of Safety Case (BSC) reports that support the PCSR.
  SFCs are structured and uniquely numbered to ensure that there is a robust and clear link between the SFC and the Fundamental Safety Function (FSF) that it supports. Hitachi-GE have defined the FSFs as:

  1. Control of reactivity
  2. Fuel cooling
  3. Long term heat removal
  4. Confinement/ Containment of radioactive materials
  5. Others (largely for support functions whose support is required for one or more of the above safety functions)

  Beneath each of these Fundamental Safety Functions a number of High Level Safety Functions (HLSFs) have been defined (e.g. Functions to maintain core geometry, Function to make up reactor coolant). All SFCs are uniquely numbered, with a numbering scheme that incorporates identifiers for the Fundamental Safety, the High Level Safety Function and the SSC. In this way, it is possible to link each SSC with the safety function it delivers.

- Safety Properties Claims (SPCs):

  SPCs are claims which justify that the UK ABWR meets Hitachi-GE's Nuclear Safety and Environmental Design Principles (NSEDPs) [Ref-22]. While SFCs describe the functions required of an SSC in order to achieve the requirements of the safety case, SPCs are system level properties such as redundancy, diversity and environmental qualification, which can help to fulfil many different safety functions.
  For this reason, SPCs are not directly linked to the HLSFs, and the HLSF number is not used as part of the unique SPC number.

- Human Based Safety Claims (HBSCs):

  HBSCs are claims on actions performed by humans to achieve safety and resilience, either in terms of maintaining normal plant state and responding to abnormal and fault events.

Further description of Hitachi-GE's approach to CAE can be found in the GDA Safety Case Development Manual (SCDM) [Ref-18], and in Chapter 5 of this PCSR.

1.4.3 Fault Fault Based and System Based View of the UK ABWR Safety Case

The PCSR contains the Safety Case for UK ABWR spread over a number of chapters. Engineering chapters describe the SSCs that make up the design and link through to Fault Studies, and Fault Studies links back to the engineering chapters. The links are provided by two main indexes: the SFCs on the engineering systems and the Fault Schedule ID for faults. These links enable the overall safety case to be followed through the PCSR starting from either the engineering chapters or Fault Studies. The process is shown diagrammatically in Figure 1.4-2

The Fault Schedule (FS) lists the faults identified for UK ABWR. Each entry has an identification number (FS ID). Chapter 24 does not describe all faults in the FS but only describes a number of bounding faults. These are shown in Table 24.4-1 with the FS ID along with the section in Chapter 24 where the description may be found and the Acceptance Criteria for the fault. The Acceptance Criteria are used to confirm that the design is tolerant to the particular bounding fault and are listed in Appendix A to Chapter 24 with a link to the specific SFC in the engineering chapters to which they relate. This link is referenced to the specific PCSR section number where the acceptance Criteria are defined and to the specific SFC in Appendix A of the relevant chapter.

Each bounding fault description in Chapter 24 identifies the HLSFs required to meet the Acceptance Criteria and the SSCs claimed in the analysis to provide those HLSFs. It also lists the faults bounded by the fault with their FS IDs. Appendix A gives a reference to the PCSR section where the SSC is described and to the specific SCF claimed and described in appendix A of the relevant chapter. The system description also describes the systems that support the claimed SSCs and the corresponding SFCs associated with them.

Thus, it is possible to start from a particular fault and trace the safety case to the engineered systems and their SFCs that are claimed to meet Acceptance Criteria.

With regard to the systems based view, the same traceability is possible through the use of HLSFs and SFCs provided in the claims tables (Appendix A). Appendix A of each chapter then gives a complete list of all faults where these SFCs are claimed along with the FS ID explained in the preceding paragraphs.

Thus, it is possible to start from a particular SSC and trace its contribution to the Safety Case by tracing where it is claimed in the fault assessment and vice versa.

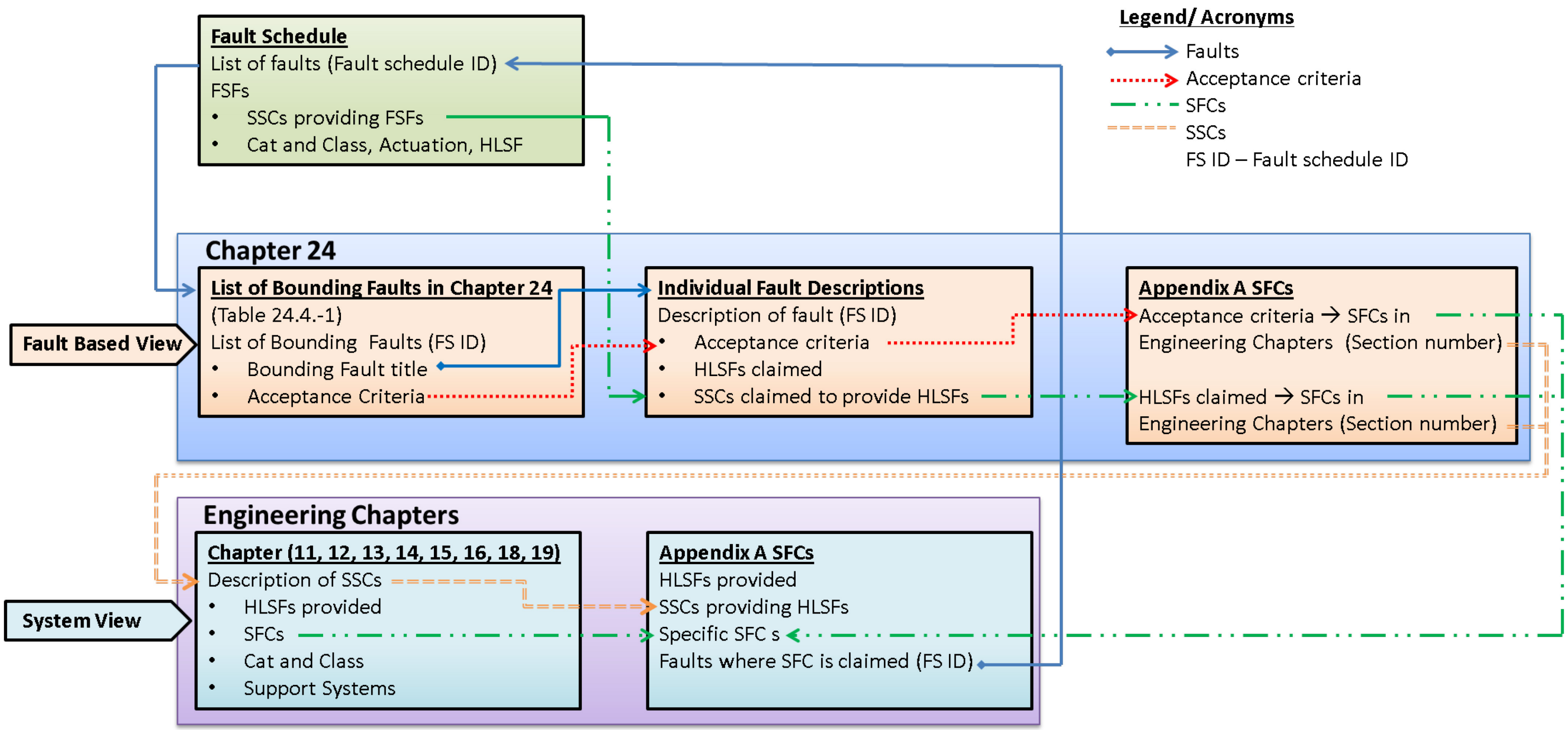

Figure 1.4-2: Schematic Presentation of Fault Based and System Based View of the UK ABWR Safety Case

1.4.4 Structure and Contents

To cover all the relevant phases of the plant cycle, the Generic PCSR contents are structured and developed with consideration of the following:

- Internationally recognized practices and guides for development of Safety Reports. Such guides include:
    - IAEA Safety Guide GS-G-4.1 [Ref-3].
- Benchmarked against previous related documentation such as
    - Generic PCSRs from previous GDAs and Sizewell B Station Safety Report (SSR); US Regulatory Guide 1.206 [Ref-4];
    - Establishment Permit Application for Nuclear Reactor Installation in Kashiwazaki-Kariwa Nuclear Power Station (Addition of Unit 6 and Unit 7) [Ref-5].
- Inclusion of additional chapters to incorporate site specific and operational matters to enable transition from GDA into site specific phase.

Based on the above approach, the UK ABWR's Generic PCSR is comprised of five parts covering all aspects of the plant lifetime.

- **Part-I: General Issues**

    Part-I sets the scene for the Generic PCSR, and describes some of the generic issues that impact on the safety of the plant (e.g. management of safety, categorisation and classification and categorisation of SSCs, definition of applicable codes and standards). This section also describes the generic site characteristics that form the basis of GDA for UK ABWR and also summarises the case that UK ABWR is resilient to internal and external hazards. Issues around Structural Integrity are also covered. Part- I comprises the following chapters:

    - Chapter 1: Introduction
    - Chapter 2: Generic Site Envelope
    - Chapter 3: Site Characteristics (Not included in Generic PCSR)
    - Chapter 4: Safety Management Throughout Plant Lifecycle
    - Chapter 5: General Design Aspects
    - Chapter 6: External Hazards
    - Chapter 7: Internal Hazards
    - Chapter 8: Structural Integrity

- **Part-II: Technical Systems**

Part-II describes the main technical systems of the UK ABWR and summarises the main functional requirements and technical specifications of SSCs required to deliver safety functions. Part- II comprises the following chapters:

- Chapter 9: General Description of the Unit (Facility)
- Chapter 10: Civil Works and Structures
- Chapter 11: Reactor Core
- Chapter 12: Reactor Coolant Systems, Reactivity Control Systems and Associated Systems
- Chapter 13: Engineered Safety Features
- Chapter 14: Control and Instrumentation
- Chapter 15: Electrical Power Supplies
- Chapter 16: Auxiliary Systems
- Chapter 17: Steam and Power Conversion Systems

- **Part-III: Systems and Processes to Support Operation, and Engineering Substantiation**

Part-III defines the auxiliary and support systems that are necessary for performance of the SSCs. It also outlines processes for management of solid, liquid and gaseous radioactive materials, to protect and reduce the risks of radioactive exposure to operators, workers and the general public. The reduction of risk will also involve identification of human-machine interfaces that are important to safe operation. Part of this section will be dedicated to preliminary frameworks on emergency preparedness, which will be developed in detail during the site specific phase. Part-III comprises of the following chapters:

- Chapter 18: Radioactive Waste Management
- Chapter 19: Fuel Storage and Handling
- Chapter 20: Radiation Protection
- Chapter 21: Human Machine Interface
- Chapter 22: Emergency Preparedness
- Chapter 23: Reactor Chemistry

- **Part-IV: Assessment**

Part-IV presents a summary of the wide ranging safety analysis and assessment work that has been used to confirm and inform the design of SSCs. These include, design basis analysis, probabilistic safety assessment, beyond design basis and severe accident analysis, and human factors evaluation. Furthermore, it also presents an overview of the ALARP case for UK ABWR, drawing on ALARP evaluations across the wide safety case. Part-IV comprises of the following chapters:

- Chapter 24: Design Basis Analysis
- Chapter 25: Probabilistic Safety Assessment

- Chapter 26: Beyond Design Basis and Severe Accident Analysis
- Chapter 27: Human Factors
- Chapter 28: ALARP Evaluation

- **Part-V: Framework of Dealing with Issues Specific to Plant Life Phase**
  Part-V describes how safety will be achieved and maintained throughout the plant's lifetime, through commissioning, operation and decommissioning, including Spent Fuel Interim Storage. Part-V comprises of the following chapters:

  - Chapter 29: Commissioning
  - Chapter 30: Operation
  - Chapter 31: Decommissioning
  - Chapter 32: Spent Fuel Interim Storage

A detailed table of contents for the whole of the Generic PCSR is presented in Chapter 0 "Generic PCSR : Master Table of Contents" in document GA91-9101-0101-00000 (XE-GD-0225) Rev. C.

While the content of each chapter is different, there are common elements that each chapter must cover (such as introduction, purpose, scope, document structure, etc.). The sub structure of each chapter has been defined to ensure consistency, and also to highlight important issues for the wider safety case such as ALARP justification, assumptions and high level limits and conditions. This generic sub structure can be summarised as:

- Introduction (including background and description of document structure)
- Purpose and Scope
- Chapter specific content*1
- Summary of ALARP Justification (for relevant topics (e.g. not for Chapter 9: General Description))
- Assumptions, Limits and Conditions for Operation (for relevant topics (e.g. not for Chapter 9: General Description))
- Document Map (showing the PCSR chapter and its key Level 2 supporting documentation)
- Table of Safety Functional Claims (SFCs) or Human Based Safety Claims (for Chapter 27) and Safety Properties Claims (SPCs) (for chapters where these have been defined)

*1. This is subdivided in to a number of sections depending on the contents of the specific PCSR chapter.

1.4.5 Scope

The scope of GDA is underpinned by three components:

- The Systems, Structures and Components (SSCs) to be included
  The extent of SSCs to be covered has been defined in line with GDA Guidance [Ref-1], and is based on a single unit at a generic site in the UK. Specifically, this includes those civil SSCs that are Safety Class 1 or 2, or Seismic Category 1 and 1A whose design is predominantly independent of the site specific conditions. These structures are listed in Generic PCSR Chapter 10. With Regard to the Systems and components, the GDA, in nature focuses on aspects of safety (for PCSR). The scope of GDA is primarily set as systems and components classified as class 1 and 2. However, as part of ensuring key aspects of lifecycle phases are covered, some of the lower class systems and components are also covered at a proportionate level of details as explained in the 'Documentation submitted' section below.

- The lifecycle phases and processes to be covered
  In line with GDA guidance [Ref-1] UK ABWR safety submissions cover (at an appropriate level of detail) all through-life aspects including design, manufacturing, construction, installation, testing, commissioning, operation, maintenance and decommissioning of the power station. In addition, the guidance requires the RP to identify the management arrangements for spent fuel and radioactive waste arising from operation of the reactors for their projected life.

- Documentation submitted
  The third point, the scope of documents, has been developed into the Design Reference for the UK ABWR [Ref-9]. However, given that GDA is a pre-licensing process aimed at risk mitigation, it is not intended to prepare a full set of design documentation for all plant lifecycle phases at the GDA stage. Furthermore, some aspects of the design and processes require significant operator input and therefore the scope of GDA documents will be limited. It is intended that 'conceptual' design information will be presented in GDA.

The eventual detailed scope of the generic UK ABWR design will be defined by the following suite of documents:

- List of Systems, Structures, and Components for Definition of UK ABWR GDA Scope [Ref-6]
- Scope of GDA document [Ref-11]
- Master Document Submission List (MDSL) [Ref-10]

Although the Generic PCSR concentrates on the safety of the ABWR design built on a generic site, it will also include arrangements to facilitate smooth transition to a specific site specific phase. To this effect:

- Hitachi-GE and Horizon Nuclear Power (HNP – who will undertake the site specific activities) have already established work streams such as multiparty discussions on scope optimization, joint workshop on Generic PCSR and Site Specific (SS) PCSR [Ref-15].

- As part of moving the GDA to the operating regime, the RP has started implementing processes for identification and capturing necessary activities and information that will eventually be transferred to the operator. For instance, the chapter level structure of the

Generic PCSR has been revised to enable authors to identify if there are any assumptions, operating limits and conditions in the corresponding Generic PCSR chapters and provide a link to where the corresponding assumptions, operating limits and conditions are recorded in the corresponding Level 2 documents [Ref-17].

The eventual detailed scope of the generic UK ABWR design is listed in the List of Systems, Structures, and Components for Definition of UK ABWR GDA Scope (this document refers to the scope of a standard ABWR design) [Ref-6] and revised Scope of GDA document (this document refers to the specific scope of the UK ABWR design) [Ref-11].

With regard to the scope of the documents, a preliminary definition has been recorded in the Master Document Submission List (MDSL) [Ref-10] and a final revision of the documents [Ref-6, 10, and 11] is expected to reflect the final position, at end of the GDA process including relevant design changes agreed during GDA.

1.5 PCSR User Guide and Route Map

Chapter 2: Generic Site Envelope

This chapter defines the Generic Site Conditions that are included within the Generic Site Envelope and will justify that the values used are robust, conservative and suitable for use in the generic design of the UK ABWR PCSR.

Chapter 3: Site Characteristics (Not included in Generic PCSR)

This chapter is not included in the Generic PCSR, and has been included to facilitate inclusion of site specific information during the site licensing phase.

Chapter 4: Safety Management throughout Plant Lifecycle

This chapter describes Hitachi-GE's generic safety management arrangements, and how these contribute to ensuring that the required levels of nuclear safety will be delivered throughout the lifetime of the UK ABWR (design, construction, commissioning, operation, and decommissioning). It describes how these safety management arrangements will support the future operator and licensee in meeting safety and environment limits and conditions, including describing the arrangements for sharing knowledge of the UK ABWR technology.

Chapter 5: General Design Aspects

Chapter 5 sets out the general principles and definitions that are used in the remainder of the PCSR, including key terminology and concepts such as Fundamental Safety Functions, High Level Safety Functions, Safety Functional Claims and Safety Properties Claims. It also describes Hitachi-GE's approach to Safety Function Categorisation and Safety Classification for SSCs. The chapter also lists the principal codes and standards used for UK ABWR and describes the principles of examination, maintenance, inspection and testing that will be applied to SSCs.

Chapter 6: External Hazards

External hazards are hazards that originate outside the site, and over which the operator has little or no control. They include natural hazards such as earthquakes, flooding and high wind; and also man made hazards such as aircraft impacts and the effects of nearby industry. Chapter 6 summarises the case that the UK ABWR will withstand external hazards without the loss of safety functions. It lists the hazards that could have an effect on safety at the plant, and demonstrates that a robust process has been used to develop this list. It also provides links to other relevant parts of the safety case that provide the justification that SSCs have been suitably designed.

Chapter 7: Internal Hazards

Internal hazards are hazards that originate on the site, and that can prevent SSCs from performing the safety functions required of them. They include such hazards as fires, explosions, flooding and dropped loads. Chapter 7 demonstrates that the risks due to internal hazards for UK ABWR have been reduced As Low As Reasonably Practicable (ALARP).

Chapter 8: Structural Integrity

Chapter 8 describes how the structural integrity of SSCs will be ensured, and summarises the evidence to substantiate the structural integrity claims made in the safety case. The chapter also introduces the structural integrity classifications of High Integrity (HI) and Very High Integrity (VHI) components and identifies the additional requirements for these components over and above those of “Standard” Class 1 SSCs.

Chapter 9: General Description of the Unit (Facility)

Chapter 9 describes the overall arrangement of the UK ABWR, and presents a high level summary description of the key systems, structures and components. It also provides links to the chapters in the Generic PCSR where detailed information is provided.

Chapter 10: Civil Works and Structures

Chapter 10 describes the civils structures that form the buildings and support structures of the UK ABWR, and specifies their safety and seismic classification. It demonstrates that the civil structures provide robust, passive protection to the nuclear safety related plant which are housed within or supported by these structures. It also presents the safety case claims (SFCs and SPCs) made on the civil structures and provides a link to supporting documents that present evidence that the claims are all achieved.

Chapter 11: Reactor Core

Chapter 11 describes the reactor core, and the nuclear fuel it will contain. It summarises the processes involved in fuel and core design and describes the characteristics of the initial core and equilibrium core designs. It also sets out the safety claims on the reactor core and fuel and demonstrates that there is an adequate level of confidence that the safe of operation of the fuel and core can be substantiated.

Chapter 12: Reactor Coolant Systems, Reactivity Control Systems and Associated Systems

Chapter 12 describes the reactor coolant systems and associated systems, which include the Reactor Pressure Vessel (RPV), Reactor Pressure Containment Boundary, and the Residual Heat Removal System. It also covers reactivity control systems such as the Control Rod Drive System (CRD) and the Standby Liquid Control System (SLC). It describes the various modes of operation of the systems and points to where the arguments and evidence that substantiate the safety claims on these systems are presented in supporting documents.

Chapter 13: Engineered Safety Features

Chapter 13 describes the principal engineered safety features of the UK ABWR, including the Emergency Core Cooling System (ECCS) and the Primary Containment Vessel (PCV). It lists the safety claims (SFCs and SPCs) made on the systems and specifies their safety classification. It also links to the supporting documents that substantiates that all relevant safety claims are adequately underpinned.

Chapter 14: Control and Instrumentation

Chapter 14 justifies that UK ABWR C&I Systems will operate with sufficient integrity to ensure that overall risks have been reduced ALARP. It presents an overview of the C&I architecture of the UK ABWR and describes the C&I platforms used, including the Class 1 Field Programmable Gate Array (FPGA) based system, Class 2 hard-wired backup system and the Class 3 control system, and how independence between these systems is achieved. It also lists the safety functions and properties (SFCs and SPCs) required to achieve safety, and demonstrates that the C&I systems will achieve these with the required level of reliability. It also describes how an appropriate design process is being followed taking into account applicable standards and the safety class of each system.

Chapter 15: Electrical Power Supplies

Chapter 15 describes the UK ABWR Electrical Power System (EPS), and justifies that it is capable of supplying power to all of the SSCs necessary to ensure safety, during both normal and fault conditions.

Chapter 16: Auxiliary Systems

Chapter 16 covers a wide range of plant systems that help to support safe operation of the plant, including the Emergency Diesel Generators (EDGs), Heating Ventilating and Air Conditioning Systems (HVAC) and process auxiliary systems. It describes the various modes of operation of the systems and justifies that they are able to meet all relevant SFCs and SPCs.

Chapter 17: Steam and Power Conversion Systems

Chapter 17 describes the systems that are used to generate electricity, which are primarily located in the Turbine Building. These systems include the Turbine Generator, the Condenser, and the mechanical components of the Turbine Electro-Hydraulic Control Systems (EHC). It points to the supporting documentation and evidence which justifies that all of the auxiliary systems achieve the requirements placed on them by the safety case.

Chapter 18: Radioactive Waste Management

Chapter 18 identifies and describes how all the waste generated during the operational phase by the station can be safely processed, stored, and dispose of to an authorised disposal site. The designs of the majority of the radioactive waste management systems are at concept design which aligns with regulatory guidance for Generic Design Assessment (GDA), and is based on proven technology. Although at concept design stage, the design is sufficiently developed to enable a high level assessment of the risks associated with radioactive waste operations. Chapter 18 demonstrates that a viable waste management strategy is available without foreclosing specific ALARP options and that the risks associated with the design and operation of the radioactive waste management systems for the UK ABWR are capable of being reduced ALARP.

Chapter 19: Fuel Storage and Handling

This chapter gives an overview of the UK ABWR fuel route, and covers receipt of new fuel, fuel handling, Spent Fuel Pool (SFP) storage and Spent Fuel Export (SFE). It sets out the safety claims for fuel storage, handling and export, and demonstrates that risks have been reduced to ALARP, by reference to supporting Topic Reports (TRs) and Basis of Safety Case (BSC) reports.

Chapter 20: Radiation Protection

This chapter describes how radiation doses to workers and the public will be controlled, and demonstrates that exposures will be ALARP. It describes how the sources of radiation have been identified and defined, and how this source term is appropriately conservative. It summarises the protection (e.g. shielding) that has been put in place to reduce direct radiation, and also describes the arrangements for contamination control and radiation monitoring. This chapter also presents the results of detailed doses assessments that estimate potential doses to workers and members of the public, and demonstrates that they are ALARP. This chapter also considers post-accident accessibility, to ensure that the required plant areas can be safely entered in the event of a range of postulated accident scenarios.

Chapter 21: Human Machine Interface

For the UK ABWR to operate safely, human operators must interact with engineering systems. This chapter describes the interfaces through which operators can influence nuclear safety, which include the Main Control Room (MCR), Remote Shutdown System (RSS) and local control stations around the UK ABWR. It describes the use of the Human Machine Interfaces (HMIs) in normal operation (including testing and maintenance) and in fault conditions. It goes on to justify how these systems meet all relevant safety requirements, including reliability requirements, usability requirements, and requirements from relevant codes and standards. This chapter is intimately linked with Chapter 27: Human Factors (HF) (which describes the analysis of human task performance) and Chapter 14: C&I (which describes the systems used to control nuclear safety related functions).

Chapter 22: Emergency Preparedness

This chapter provides a high level overview of the generic emergency preparedness arrangements applicable to the UK ABWR. It describes the emergency facilities, vehicles and equipment appropriate for the UK ABWR, but does not propose any specific technology, as this is a site/ operator specific decision. The chapter also outlines how the on site response organisation would interact with national, regional and local response organisations.

Chapter 23: Reactor Chemistry

Reactor chemistry influences many aspects of the plant including structural integrity, radioactive source term and off-site discharges. This chapter describes the UK ABWR chemistry regime and justifies why it is in line with relevant good practice and is optimised when considering all factors, e.g. SSC degradation mechanisms, radiological dose implications and waste management requirements.

Chapter 24: Design Basis Analysis

Chapter 24 introduces the fault studies for the UK ABWR, and describes how faults have been identified and how they have been grouped into bounding faults for detailed assessment. These bounding faults are the starting point for the UK ABWR fault schedule, which is a fundamental document for the safety case that links potential fault scenarios with the systems that protect against them. The chapter describes how the fault schedule has been developed, identifies those faults that are considered to be within the design basis, and summarises the assessment of each of the faults identified. The chapter confirms that the UK ABWR SSCs adequately protect against or mitigate the consequences of all DB faults, and that those SSC are classified appropriately.

Chapter 25: Probabilistic Safety Assessment

Probabilistic Safety Assessment (PSA) is a key tool to assess plant risks, identify potential plant vulnerabilities and inform ALARP decision making. Chapter 25 summarises the methodology and the quantification results for a full scope PSA carried out to support GDA. The scope of the PSA covers at-power operation, shutdown, spent fuel pool and other non-reactor faults. The PSA also considers external hazards and internal hazards. The chapter confirms that the relevant risk targets have largely been met. A summary is also provided on the use of PSA to support ALARP demonstration and the identification of any reasonably practicable design improvements.

Chapter 26: Beyond Design Basis and Severe Accident Analysis

Beyond Design Basis Analysis (BDBA) identifies and analyses events that have a lower frequency than Design Basis accidents, usually as a result of multiple failures or common cause failures of protection systems. Severe accidents are defined as those fault sequences that could potentially lead to large societal effects and a significant number of deaths off-site.
Chapter 26 extend the DB analysis of Chapter 24 to beyond design basis events and severe accidents, and provides inputs to the PSA (Chapter 25) and the emergency arrangements discussed in Chapter 22. A summary is provided on the scope, methods and results of analysis. A summary is also provided on the ALARP assessments for a number of severe accident mitigation features in the design and it concludes that these features are consistent with Relevant Good Practice.

Chapter 27: Human Factors

This chapter provides an overview of the Human Factors (HF) work that has formed an integral part of the UK ABWR GDA project. It also summarises the HF aspects of the UK ABWR safety case, describing the structure of the HF supporting documentation, and showing where greater detail on the claims, arguments and evidence is presented. The chapter introduces the Human-Based Safety Claims (HBSCs) that have been used to support the achievability of the SFCs and SPCs made throughout the Generic PCSR. It summarises the processes used to identify the specific HBSCs and demonstrate that they can be achieved. This chapter concludes that there has been, and will continue to be, adequate and timely impact of HF on the UK ABWR design, which will ensure that it supports the human performance expected in the safety case.

Chapter 28: ALARP Evaluation

Chapter 28 demonstrates that the UK ABWR meets the ALARP principle. It does this in two ways. Firstly, it describes how the UK ABWR has evolved over time and how, with each successive evolution from the earliest BWRs to the present, the design has increased the level of safety and reduced risk. Secondly, the chapter describes how the ALARP principle has been applied during the design of the UK ABWR, referring to probabilistic and optioneering studies that have been used to inform the final design.

Chapter 29: Commissioning

Commissioning is an essential process for the subsequent safe operation of a nuclear power plant and it should be carefully developed, planned and executed. Chapter 29 describes the processes that should be followed to ensure that the UK ABWR will be safely commissioned. These include principles and strategy of commissioning, the generic plan for commissioning and how the hazards and risks in different stages of commissioning will be controlled and reduced ALARP.

Chapter 30: Operation

Chapter 30 describes the generic approach to operations required for safely operating a UK ABWR. It includes discussion of principles, programmes, processes, operating arrangements, organisation, roles and responsibilities. The description of the concept of operations is necessarily generic during GDA, but it has been developed in collaboration with a potential future site licensee to ensure alignment with current UK modern nuclear industry good practice for operations.

Chapter 31: Decommissioning

Decommissioning is the last stage in the overall lifecycle of a facility, but it must be taken into account at the planning and design stages so that appropriate steps are taken to prevent or mitigate potential decommissioning challenges and risks. This chapter demonstrates that the UK ABWR is capable of being decommissioned safely, with risks reduced ALARP.

Chapter 32: Spent Fuel Interim Storage

The spent fuel created by the UK ABWR will initially be stored in the spent fuel pool inside the reactor building. After a number of years it will be transferred to a Spent Fuel Interim Storage (SFIS) facility where it will be stored until final disposal off-site is possible. The final design of the spent fuel storage system will not be required until significantly later than GDA, and will be decided on by the future licensee. The purpose of Chapter 32 is to demonstrate how the 'concept' SFIS assumed for GDA is capable of reducing risks ALARP and confirm that a viable SFIS strategy is available to future licensees without foreclosing specific options.

The roles of each of the chapters in the PCSR, and the summaries of their contents are provided in the Executive Summaries included in each chapter.

1.6 Safety Assessments and Achievements of the ABWR Design Prior to the UK ABWR

The ABWR design was developed based on extensive construction and operating experience of Boiling Water Reactor (BWR) plants in Japan, United States of America (USA) and Europe, and was jointly developed by the BWR suppliers with support from BWR operators. The ABWR is the only generation III+ [Ref-7] plant that has been in commercial operation for over 15 year. The design has been independently assessed, certified and /or licensed by Regulators in Japan [Ref-5], Taiwan and in the USA [Ref-8].

The first and the second ABWR plants, which are both owned and operated by the Tokyo Electric Power Company (TEPCO), were constructed at the Kashiwazaki-Kariwa Nuclear Power Station in Japan and commenced commercial operation in 1996 and in 1997, respectively. Hitachi-GE was involved in the development, design, construction, and commissioning of both of these ABWRs.

Since then, there has been two more successful ABWR design, and construction projects in Japan, both of which Hitachi-GE was engaged. There are three further ABWR plants under construction in Japan. Hitachi-GE has been engaged in the design and construction of all seven ABWR plants in Japan, and has accumulated significant experience in both design and construction of the ABWR.

Hitachi-GE is confident that the present Japanese ABWR design serves as a technical baseline, with modifications made as necessary to meet UK requirements and criteria.

1.7 Abbreviations and Acronyms List

The commonly used terminologies, acronyms and abbreviations in the UK ABWR design and the GDA process are shown in Appendix A.

NOT PROTECTIVELY MARKED

1.8 Conclusions

This chapter defines the top level claims of the UK ABWR design and provides an overview of the Generic PCSR and high level links to the relevant information that supports Hitachi-GE's top claim that the UK ABWR can be safety operated in the UK. It provides high level claims that underpin this top claim. Chapter 1 provides the key achievements of the UK ABWR in Steps 1, 2, 3 and 4.

This chapter provides a guide to the structure of the PCSR and provides summaries of the contents of each of the chapters. The chapter also describes the key approach adopted in the development of UK ABWR safety cases and provides references to the relevant GDA documentation.

1.9 References

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Appendix A: Abbreviations and Acronyms List

	3D-CAD	3D-Computer Aided Design
	3D-CAE	3D-Computer Aided Engineering
A	A/E	Architect Engineer
	AAC	Alternate AC
	ABS	Absolute
	ABWR	Advanced Boiling Water Reactor
	AC	Alternating Current
	AC	Atmospheric Control system
	ACI	American Concrete Institute
	ACIWA	AC - Independent Water Addition
	ACoP	Accepted Code of Practice
	ACP	Access Control Point
	ACRS	Advisory Committee on Reactor Safety
	ACS	Reactor / Turbine Auxiliary Control System
	ActP	Actinide Product
	ACU	Air Conditioning Unit
	AD/B	Administration Building
	ADS	Approved Dosimetry Service
	ADS	Automatic Depressurisation System
	AEC	Atomic Energy Commission (US, JP)
	AEC	Atomic Energy Council (Taiwan)
	AEC	Auxiliary Equipment Control system
	AECC	Alternative Emergency Control Centre
	AEOD	Office of Analysis and Evaluation of Operational Data
	AET	Advanced Engineering Team
	AFC	Air Fin Cooler
	AFC	Automatic Frequency Control
	AFIP	Automatic Fixed In-Core Probe
	AFW	Auxiliary Feedwater
	AHEF	Alternate Heat Exchange Facility
	AHU	Air Handling Unit
	AI	Aircraft Impact
	AIA	Airplane Impact Assessment
	AIJ	Architectural Institute of Japan
	AISC	American Institute of Steel Construction
	ALAP	As Low As Practicable
	ALARA	As Low As Reasonably Achievable
	ALARP	As Low As Reasonably Practicable

Appendix A: Abbreviations and Acronyms List

ALF	Automated Load Following
ALWR	Advanced Light Water Reactor
AM	Accident Management
AMG	Accident Management Guidelines
AMP	Aging Management Programme
ANI	Alternative Nitrogen Injection System
ANS	American Nuclear Society (US)
ANSI	American National Standards Institute (US)
ANT	Auxiliary Normal Transformer
AO	Air Off Take System
AOF	Allocation of Function
AOFR	Allocation of Function Report
AOO	Anticipated Operational Occurrences
AOP	Abnormal Operating Procedure
AOV	Air Operated Valve
AP	Activation Product
APC	Airplane Crash
APD	Alarm Pocket Dosimeter
API	American Petroleum Institute
APLHGR	Average Planar Linear Heat Generation Rate
APR	Automatic Power Regulator System
APRM	Average Power Range Monitor
ARD	Anti-Reverse Rotation Device
ARI	Alternative Rod Insertion
ARM	Area Radiation Monitoring System
ARMC	Automated Rod Movement Control
ARS	Acceleration Response Spectrum
AS	Turbine Auxiliary Steam System
ASCE	American Society of Civil Engineers
ASD	Adjustable Speed Drive
ASHRAE	American Society of Heating, Refrigerating and Air-conditioning Engineers, Inc
ASI	Adverse System Interactions
ASIC	Application Specific Integrated Circuits
ASL	Assumed System Loss
ASME	American Society of Mechanical Engineers
AST	Auxiliary Standby Transformer
ASTM	American Society for Testing and Materials
ATIP	Automatic Traversing Incore Probe
ATLM	Automated Thermal Limit Monitor

	ATWS	Anticipated Transients Without Scram
	ATWSI	Anticipated Transient Without Scram Instability
	ATWT	Anticipated Transient Without Trip
	AUXB	Auxiliary Boiler
	AUXRP	Auxiliary Room Panel
	AVR	Automatic Voltage Regulator
	AWJC	Abrasive Water Jet Cutting
	AWS	American Welding Society
	AWWA	American Water Works Association
B	B&PV	Boiler and Pressure Vessel (ASME Code)
	B/B	Back-up Building
	BA	Breathing Air System
	BA	Breathing Apparatus
	BAC	Bead Activated Carbon
	BAR	Baseline Human Factors Assessment Report
	BAT	Best Available Technique
	BBCP	Back-up Building Control Panel
	BBCR	Back-up Building Control Panel Room
	BBECR	Backup Building Emergency Control Room
	BBEE	Backup Building Electrical Equipment
	BBEE/Z	Backup Building Electrical Equipment Zone
	BBG	Back-up Building Generator System
	BBGFO	Back-up Building Generator Fuel Oil System
	BBTS	Backup Building Transfer Switch
	BCCS	Back-up Canister Cooling System
	BCP	Ball Circulation Pump
	BDB	Beyond Design Basis
	BDBA	Beyond Design Basis Analysis/Accident/Assessment
	BDL	Bottom Drain Line
	BF	Boiler Feed
	BFP	Boiler Feed Pump
	BFPT	Boiler Feed Pump Turbine
	BFW	Boiler Feedwater
	BGS	British Geological Survey
	BHP	Brake Horsepower
	BLDG	Building
	BOC	Bottom of Core
	BOC	Beginning of Cycle

Appendix A: Abbreviations and Acronyms List

	BOP	Balance of Plant
	BPMS	Banked Position Withdrawal Sequence
	BPU	Bypass Unit
	BPVC	Boiler and Pressure Vessel Code
	BS	British Standard
	BSC	Basis of Safety Case
	BSL	Basic Safety Level
	BSO	Basic Safety Objective
	BT	Boiling Transition
	BTC	BWR Training Centre (JP)
	BTP	Branch Technical Position
	BWR	Boiling Water Reactor
	BWROG	BWR Owners' Group (US)
	BWRT	Backwash Receiving Tank
C	C&C	Categorisation & Classification
	C&I	Control and Instrumentation
	C/B	Control Building
	C/C	Cooling Coil
	CAD	Controlled Area Drain System
	CAD	Computer Aided Design
	CAE	Claim-Argument-Evidence
	CAE	Computer Aided Engineering
	CAMS	Containment Atmospheric Monitoring System
	CAP	Cargo Access Portal
	Cat	Category
	CAV	Cumulative Absolute Velocity
	CB	Circuit Breaker
	CBA	Cost Benefit Analysis
	CBC2EE/Z	Control Building Class 2 Electrical Equipment Zone
	CBEEE	Control Building Electrical Equipment
	CC	Condenser Tube Cleaning System
	CCDF	Complimentary Cumulative Distribution Failure
	CCF	Common Cause Failure
	CCFP	Conditional Containment Failure Probability
	CCI	Core-Concrete Interaction
	CCL	Cable & Conduit List
	CCS	Containment Cooling System
	CCS	Canister Cooling System

Appendix A: Abbreviations and Acronyms List

CCTV	Closed-Circuit Television
CCTV	Remotely controlled television
CD	Condensate Demineraliser System
CDA	Cask Drop Accident
CDF	Core Damage Frequency
CDG regulations	Carriage of Dangerous Goods and Use of Transportable Pressure Equipment Regulations
CDM	Construction(Design and Management)
CDRL	Core Damage Radiation Level
CERT	Constant Extension Rate Test
CET	Containment Event Tree
CF	Condensate Filter System
CFDW	Condensate and Feedwater System
CFM	Core Flow Measurement Systems
CFR	Code of Federal Regulation
CGCS	Combustible Gas Control System
CHF	Critical Heat Flux
CHFR	Critical Heat Flux Ratio
CID	Criticality Incident Detection system
CILC	Crud Induced Localised Corrosion
CIS	Chemical Injection System
CIV	Combined Intermediate Valve
CLAB	Spent Fuel Storage Facility in Switzerland
CLI	Criteria for Limiting Impact
CM	Configuration Management
CMPF	Common Mode Probabilistic Failure
CMU	Control Room Multiplexing Unit
CNRA	Committee on Nuclear Regulatory Activities (of the OECD-NEA)
CNS	Civil Nuclear Security (part of the Office for Nuclear Regulation) (UK)
CO	Condensation Oscillation
COD	Commercial Operation Date
COM	Communication System
COMAH	Control of Major Accident Hazards Regulations of 1999
CONW	Concentrated Waste System
COPS	Containment Overpressure Protection System
COR	Concept of Operations Report
CORD	Chemical Oxidation Reduction Decontamination
CP	Core Plate
CP	Construction Permit (JP)

Appendix A: Abbreviations and Acronyms List

	CP	Control Panel
	CP	Corrosion Product
	CPDP	Core Plate Differential Pressure
	CPED(A)	Chemical / Process Engineering Design (Approach)
	CPLD	Complex Programmable Logic Device
	CPM	Chief Project Manager (Hitachi-GE)
	CPR	Critical Power Ratio
	CPS	Condensate Purification System
	CPU	Central Processing Unit
	CR	Control Rod
	CRD	Control Rod Drive (System)
	CRGT	Control Rod Guide Tube
	CRO	Control Room Operator
	CRS	Control Room Supervisor
	CRT	Cathode Ray Tube
	CRUD	Chalk River Un-identified Deposit
	CS	Containment Spray
	CS	Control Switch
	CS	Computer System
	CSA	Conceptual Security Arrangements
	CSC	Company Support Centre
	CST	Condensate Storage Tank
	CT	Cask Transporter
	CT	Current Transformer
	CTP	Condensate Transfer Pump
	CUW	Reactor Water Clean-up System
	CV	(Turbine) Control Valve
	CVCF	Constant Voltage Constant Frequency
	CW	Circulating Water System
	CW/S	Circulating Water Structure
	CWP	Circulating Water Pump
D	D/F	Diaphragm Floor
	D/P	Differential Pressure
	D/S	Dryer/Separator
	D/W	Drywell
	DA	Design Authority
	DAC	Design Acceptance Confirmation
	DAG	Diverse Additional Generator

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DAP	Duly Authorised Person
DAS	Data Acquisition System
DAW	Dry Active Waste
DB	Ductbank
dB	Decibel
DB	Design Basis
DBA	Design Basis Analysis, Accident, Assessment
DBC	Design Basis Conditions
DBE	Design Basis Event
DBE	Design Basis Earthquake
DBG	Double Blade Guide
DBT	Design Basis Threat
DC	Design Certification (US)
DC	Direct Current
DCD	Design Control Document (US)
DCH	Direct Containment Heating
DCHV	Document Control Centre HVAC System
DCIS	Distributed Control Information System
DCN	Design Change Notice
DCO	Development Consent Order (UK)
DD	Decontamination Drain
DDF	Depth Duration Frequency
DDFP	Diesel Driven Fire Pump
DDN	Design Difference Notice
DE	Decontamination System
DECC	Department for Energy and Climate Change
DEDP	Decommissioning Developed Principles
DEFRA	Department for Environment , Flood & Rural Affairs
DEGB	Double-Ended Guillotine Break
DEPSS	Drywell Equipment and Piping Support Structure
DER	Design and Engineering Report
DF	Decontamination Factor
DFC	Damaged Fuel Container
DfD	Decontamination for Decommissioning
DG	Diesel Generator
DGAE	DG Air Intake And Exhaust System
DGCW	DG Cooling Water System
DGEE	Diesel Generator Electrical Equipment
DGEE/Z	Emergency D/G Electrical Equipment Zone

	DGFO	DG Fuel Oil System
	DGLO	DG Lubricant Oil System
	DiD	Defence in Depth
	DIV	Division
	DJR	Design Justification Report
	DMC	Digital Measurement and Control
	DOD	United States Department of Defence (US)
	DOE	United States Department of Energy (US)
	DOF	Degree of Freedom
	DOI	Dedicated Operator Interface
	DOP	Dioctyl Phthalate
	DPC	Double Packing Clearance
	DPMO	Decommissioning Project Management Organisation
	DQR	Dynamic Qualification Report
	DR	Decay Ratio
	DR	Design Reference
	DRM	Dust Radiation Monitoring System
	DRP	Design Reference Point
	DSA	Deterministic Safety Analysis
	DSEAR	Dangerous Substances and Explosive Atmospheres Regulations
	DSP	Steam Dryer, Steam Separator Pit
	DST	Deposit Source Term
	DTA	Defect Tolerance Assessment
	DTM	Digital Trip Module
	DW	Domestic Water System
	DWC	Drywell Cooling System
	DWL	Dump Water Level
	DWMF	Decommissioning Waste Management Facility
	DWMP	Decommissioning and Waste Management Plan
	DZO	Depleted Zinc Oxide
E	E/C	Erosion/Corrosion
	EAB	Exclusion Area Boundary
	EBWR	Experimental Boiling Water Reactor
	EC	European Commission
	EC	Emergency Controller
	ECC	Emergency Control Centre
	ECCS	Emergency Core Cooling System
	ECLL	Electric Room Combustible Loading Limit

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ECP	Engineering Computer Program
ECP	Electrochemical Corrosion Potential
ECR	Equivalent Cladding Reacted
EDF	Electricite de France
EDG	Emergency Diesel Generator System
EDG/B	Emergency Diesel Generator Building
EDRMS	Horizon Nuclear Power Electronic Document and Records Management System
EE	Electorical Engineering
EECW	Emergency Equipment Cooling Water System
EEMUA	Engineering Equipment & Materials Users' Association
EFPD	Effective Full Power Day
EHC	Turbine Electro-Hydraulic Control System
EIA	Environmental Impact Assessment
EIADR	Environmental Impact Assessment for Decommissioning Regulations
EL	Elevation
ELEV	Elevator
EMC	Electromagnetic Compatibility
EMI	Electromagnetic Interference
EMIT	Examination, Maintenance, Inspection and Testing
EMR	Electricity Market Reform
EMS	Essential Multiplexing System
EN	European Norm
EN	National Policy Statement for Energy
ENIQ	European Network for Inspection and Qualification
ENSREG	European Nuclear Safety Regulators Group
EOC	End Of Cycle
EOC	Errors of Commission
EOEC	End of Equilibrium Cycle
EOF	Emergency Operations Facility
EoG	End of Generation
EOL	End of Life
EOP	Emergency Operating Procedure
EOP	Main Turbine Emergency Oil Pump
EP	Establishment Permit (JP)
EPA	Environmental Protection Agency
EPC	Engineering Procurement and Construction
EPD	Electric Power Distribution System
EPFM	Elastic-Plastic Fracture Mechanics
EPG	Emergency Procedure Guideline

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	EPR	European Pressurised Reactor (Evolutionary Power Reactor)
	EPRI	Electric Power Research Institute
	EPS	Electrical Power System
	EPU	Extended Power Uprate
	EPZ	Emergency Planning Zone
	EQ	Environmental Qualification
	ERICP	Emergency Rod Insertion Control Panel
	ERIP	Emergency Rod Insertion Panel
	ERT	Emergency Response Team
	ES	Extraction Steam System
	ESBWR	Economic Simplified Boiling Water Reactor
	ESF	Engineered Safety Feature
	ESS	Essential Service System
	ETA	Event Tree Analysis
	EUR	European Utility Requirements
	EURATOM	European Atomic Energy Community
	EVA	Extreme Value Analysis
	EW	Early Works
	EXCT	Excitation System
F	F/D	Filter-Demineraliser
	FA	Fuel Assembly
	FAC	Flow Accelerated Corrosion
	FAD	Failure Assessment Diagram
	FAI	Fail As Is
	FAP	Funding Arrangement Plan
	FATT	Fracture Appearance Transition Temperature
	FBD	Function Block Diagram
	FC	Fail Close
	FCC	Fuel Cycle Costs
	FCC	Fuel Cask Cleaning Facility
	FCI	Fuel Coolant Interaction
	FCP	Forward Control Points
	FCS	Flammability Gas Control System
	FCU	Fan Coil Unit
	FCV	Flow Control Valve
	FCVS	Filtered Containment Venting System
	FD	Flat Display
	FDA	Final Design Approval (US)

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FDP	Funded Decommissioning Programme
FDW	Feedwater System
FDWC	Feedwater Control System
FE	Foreseeable Event
FEED	Front End Engineering and Design
FEH	Flood Estimate Handbook
FEI	Fe Ion Injection System
FEM	Finite Element Method
FEP	Front End Procurement
FEPC	The Federation of Electric Power Companies of Japan (JP)
FF	Fresh Fuel
FF	Frequent Fault
FHA	Fuel Handling Accident
FHA	Fire Hazard Analysis
FHD	Forced Helium Dehydration
FHM	Fuel Handling Machine
FID	Final Investment Decision
FIV	Flow-Induced Vibration
FL	Fuel Loading
FLD	Floor Leakage Detection System
FLR	Full-Length Rod
FLS	Flooding System
FLSR	Flooding System of Reactor Building
FLSS	Flooding System of Specific Safety Facility
FLT	Fork Lift Truck
FMCRD	Fine Motion Control Rod Drive
FMCRDM	Fine Motion Control Rod Drive Mechanism
FMDC	Fine Motion Driver Cabinet
FMEA	Failure Modes and Effects Analysis
FMECA	Failure Modes and Effects Criticality Analysis
FN	Ferrite Number
FO	Fail Open
FO	Field Operator
FP	Fire Protection System
FP	Fission Product
FPC	Fire Protection Code
FPC	Fuel Pool Cooling and Clean-up System
FPCMs	Fuel Pool Cooling, Clean-up and Make-up Systems
FPGA	Field Programmable Gate Array

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	FPLG	Fusible Plug
	FPM	Fuel Preparation Machine
	FPR	Fuel Pool Racks
	FPS	Freeze Protection System
	FRS	Floor Response Spectra
	FRT	Fault Ride Through
	FS	Fault Study
	FS	Fuel Support
	FS	Field Switch
	FSAR	Final Safety Analysis Report (US)
	FSER	(The ABWR) Final Safety Evaluation Report (US)
	FSF	Fundamental Safety Function(s)
	FSM	Frequency Sensitive Mode
	FSR	Flood Studies Report
	FTDC	Fault-Tolerant Digital Controller
	FV/B	Filter Vent Building
	FVI	Fussell-Vesely Importance
	FWH	Feedwater Heater
	FWHD	Feedwater Heater and Drain System
	FWLB	Feedwater Line Break
	FWRB	Feedwater Runback
G	GAC	Granular Activated Carbon
	GAHV	General Administration Building HVAC System
	GBq	Giga Becquerels
	GC	Grid Code
	GCS	Generator Cooling System
	GDA	Generic Design Assessment
	GDC	General Design Criterion (US)
	GDF	Geological Disposal Facility
	GDS	Generator Disconnecting Switch
	GE	General Electric Company
	GEH	GE Hitachi Nuclear Energy
	GEN	Generator
	GEP	Generic Environmental Permit
	GEP-RSR	Generic Environmental Permit - Radioactive Substances Regulation
	GETAB	General Electric Thermal Analysis Basis
	GGC	Generator Gas Control System
	GI	Ground Investigation

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	GIS	Geographical Information System
	GL	Ground Level
	GLS	Generator Load Switch
	GND	Ground
	GNF	Global Nuclear Fuel
	GP	Good Practice
	GSC	Generator Stator Cooling System
	GSC	Gland Steam Condenser
	GSE	Gland Steam Evaporator
	GSE	Generic Site Envelope
	GSE-DT	Gland Steam Evaporator Drain Tank
	GSEXH	Gland Steam Exhauster
	GSO	Generator Sealing Oil System
	GT	Generator Transformer
	GTS	Generic Technical Specifications
H	HSI	Human-System Interface
	HAW	Higher Activity Radioactive Waste
	HAW	Highly Activated Waste
	HAZ	Heat-Affected Zone
	HAZID	Hazard Identification
	HAZID	Hazard Identification Study
	HAZOP	Hazard and Operability (Studies)
	HAZOP	Hazard and Operability process
	HB	House Boiler
	HB/B	House Boiler Building
	HBSC	Human Based Safety Claim
	HCU	Hydraulic Control Unit
	HCW	High Chemical Impurities Waste System
	HD	Feedwater Heater Drain System
	HDL	Hardware Description Languages
	HECW	HVAC Emergency Cooling Water System
	HELB	High-Energy Line Break
	HELSA	High-Energy Line-Separation Analysis
	HEM	Homogeneous Equilibrium Model
	HEP	Human Error Probability
	HEPA	High Efficiency Particulate Air Filter
	HEX	Heat Exchanger
	HF	Human Factors

HFAR	Human Factors Assessment Report
HFE	Human Factors Engineering
HFF	Hollow Fibber Filter
HFI	Human Factors Integration
HFIP	Human Factors Integration Plan
HFIR	Human Factors Issues Register
HFMP	Human Factors Methodology Plan
HGC	Hydrogen Gas Cooling System
HHISO	Half Height International Standards Organization
HI	Hydrogen Iodide
HI	High Integrity
HIACS	microprocessor platform
HIC	High Integrity Containers
HIRE	Hazard Identification and Risk Evaluation
Hitachi-GE	Hitachi-GE Nuclear Energy, Ltd.
HLND	Hot Laundry Equipment
HLSF	High Level Safety Function(s)
HLW	High Level Waste
HMG	Her Majesty's Government
HMI	Human-Machine Interface
HMIS	Human Machine Interface System
HMS	Horizon Management System
HNCW	HVAC Normal Cooling Water System
HNP	Horizon Nuclear Power
HOIS	Hydrogen and Oxygen Injection System
HOT	Heavy Oil Tank
HP	High Pressure
HPCF	High Pressure Core Flooder System
HPCI	High Pressure Core Injection
HPCP	High Pressure Condensate Pump
HPCS	High Pressure Core Spray
HPDP	High Pressure Drain Pump
HPDT	High Pressure Drain Tank
HPIN	High Pressure Nitrogen Gas Supply System
HPPD	High Pressure Pumped Drain
HP-T	High Pressure Turbine
HRA	Human Reliability Analysis
HRAR	Human Reliability Analysis Report
HS	Heating Steam System

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	HSCR	Heating Steam and Condensate Water Return System
	HSD	Hot Shower Drain
	HSE	Health and Safety Executive (UK)
	HSO	Generator Hydrogen Seal Oil System
	HSSS	Hardware/Software System Specification
	HSWA74	Health and Safety at Work etc. Act 1974
	HV	Feedwater Heater Vent System
	HVAC	Heating Ventilating and Air Conditioning System
	HVD	Miscellaneous Heater, Drain and Vent System
	HVG	High Value Gate
	HVH	Heating Ventilating Handling Unit
	HWBP	Hardwired Back-up Panel
	HWBS	Hardwired Back-up System
	HWC	Hydrogen Water Chemistry
	HWL	High Water Level
	Hx	Heat Exchanger
	Hx/B	Heat Exchanger Building
	Hx/B-E HVAC	Heat Exchanger Building Emergency HVAC
	Hx/B-N HVAC	Heat Exchanger Building Normal HVAC
I	I&C	Instrumentation and Control
	IA	Instrument Air System
	IAEA	International Atomic Energy Agency
	IASCC	Irradiation Assisted Stress Corrosion Cracking
	IBC	International Bulk Container
	IBD	Interlock Block Diagram
	IC	Isolation Condenser
	IC	Intelligent Customer
	ICBM	Independent Confidence Building Measures
	ICC	Inadequate Core Cooling
	ICD	Interface Control Diagram
	ICEA	Insulated Cable Engineer Association
	ICGT	In-Core Guide Tube
	ICLP	International Commission on Radiological Protection
	ICM	In-Core Monitor
	ICMS	Integrated Construction Management System
	ICP	Instrument and Control Power Supply
	ICP	Initial Commissioning Program
	ICRP	International Commission on Radiological Protection

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ICS	Integrated Control System
IDAC	Interim Design Acceptance Confirmation
IDCOR	Industry Degraded Core Rulemaking
IE	Inspection and Enforcement
IEC	International Electrotechnical Commission
IED	Instrument Elementary Diagram
IEEE	Institute of Electrical and Electronics Engineers
IF	Infrequent Fault
IFV	Independent Fuel Verifier
IGBT	Insulated Gate Bipolar Transistor
IGSCC	Intergranular Stress Corrosion Cracking
IH	Internal Hazard
IHSI	Induction Heating Stress Improvement
ILRT	Integrated Leak Rate Test
ILW	Intermediate Level Waste
ILWISF	Intermediate Level Waste Interim Storage Facility
IMS	Information Management System
IN	Information Notice
INL	Instrumentation Lists
INPO	Institute of Nuclear Power Operations
INRA	International Nuclear Regulators Association
INSAG	International Nuclear Safety Advisory Group (IAEA)
INSR	Independent Nuclear Safety Review
INST	Instrumentation
IoF	Incredibility Of Failure
IOT	Infrequent Operational Transients
IPB	Isolated Phase Bus
IPC	Infrastructure Planning Commission
IRM	Intermediate Range Monitor
IRR	Ionising Radiations Regulations
IRRS	Integrated Regulatory Review Service (IAEA)
ISA	Instrument Society of America
ISF	Intake Screen Cleaning Facility
ISI	Inservice Inspection
ISLOCA	Interfacing System LOCA
ISMA	Independent Support Motion Response Spectrum Analysis
ISO	International Organization for Standardization
IST	Inservice Testing
ISV	Intermediate Steam Stop Valve

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	ITP	Inspection and Test Plan
	ITP	Initial Test Program
	ITV	Industrial Television Facilities
	IV	Intercept Valve
J	J-ABWR	Japanese Advanced Boiling Water Reactor
	J-BWROG	BWR Owners' Group (JP)
	JANTI	Japan Nuclear Technology Institute
	JEAC	Japan Electric Association Code
	JEAG	Japan Electric Association Guide
	JEC	Japanese- Electrotechnical Committee (JP)
	JEM	Japanese Electrical Manufactures (JP)
	JIS	Japanese Industrial Standard (JP)
	JNES	Japan Nuclear Energy Safety Organization (JP)
	JPO	Joint Programme Office (UK)
	JSME	The Japan Society of Mechanical Engineers
K	KAG	Key Assumptions and Ground rules
	KK	Kashiwazaki-Kariwa Nuclear Power Station
	KK-6	Kashiwazaki-Kariwa Nuclear Power Station Unit 6
	KK-7	Kashiwazaki-Kariwa Nuclear Power Station Unit 7
L	L/D	Lower Drywell
	LBB	Leak-Before-Break
	LCB	Local Control Box
	LCM	Low Cobalt Material
	LCO	Limiting Condition for Operation
	LCP	Local Control Panels
	LCV	Level Control Valve
	LCW	Low Chemical Impurities Waste System
	LD	Laundry Drain System
	LD	Load Driver
	LDF	Lower Drywell Flooder System
	LDS	Leak Detection System
	LED	Light Emitting Diode
	LEFM	Linear Elastic Fracture Mechanics
	LEMP	Lightning Electro-Magnetic Pulse
	LER	Licensing Event Report
	LERE	Licensing Event Report Evaluation

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LFCV	Low Flow Control Valve
LFE	Learning from Experience
LFSM	Limited Frequency Sensitive Mode
LHGR	Linear Heat Generation Ratio
LLC	Limited Liability Company
LLI	Long Lead Item
LLISP	Long Lead Item Safety Pack
LLW	Low Level Waste
LLWR	Low Level Waste Repository
LNTP	Limited Notice To Proceed
LO	Turbine Lubilicating Oil System
LOCA	Loss of Coolant Accident
LOMI	Low Oxidation State Metal Ion
LOOP	Loss of Off-site Power
LOPA	Loss of Power Accident
LOT	Light Oil Tank
LP	Low Pressure
LPCF	Low Pressure Core Flooder
LPCI	Low Pressure Coolant Injection
LPCP	Low Pressure Condensate Pump
LPCRD	Locking Piston Control Rod Drive
LPCS	Low Pressure Core Spray
LPDP	Low Pressure Drain Pump
LPDT	Low Pressure Drain Tank
LPFL	Low Pressure Core Flooder System
LPPD	Low Pressure Pumped Drain System
LPRM	Local Power Range Monitor
LP-T	Low Pressure Turbine
LPT	Low Profile Transporter
LPZ	Low Population Zone
LRB	Licensing Review Bases
LRF	Large Release Frequency
LRW	Liquid Radwaste System
LSP	Lighting and Servicing Power Supply
LTA	Lead Test Assemblies
LTNC	Low Temperature NobleChem™
LTP	Lower Tie Plate
LUHS	Loss of Ultimate Heat Sink
LV	Low Voltage

	LVDT	Linear Variable Differential Transformers
	LWL	Low Water Level
	LWMS	Liquid Waste Management System
	LWR	Light Water Reactor
M	M&TE	Measuring & Test Equipment
	M/C	Metal-Clad Switchgear
	M/D-RFP	Motor Driven Reactor Feedwater Pump
	MAAP	Modular Accident Analysis Program
	MAI	Manufacturing Acceptance Inspections
	MAM	Mobile Accident Management Facility
	MAPLHGR	Maximum Average Planar Linear Heat Generation Rate
	MBA	Misplaced Bundle Accident
	MBC	Media Briefing Centre
	MBP	Media Briefing Point and Marshalling Point
	MC	Metal Containment
	MCC	Motor Control Centre
	MCC	Main Control Console
	MCCI	Molten Core Concrete Interaction
	MCPR	Minimum Critical Power Ratio
	MCR	Main Control Room
	MCRS	Main Control Room Supervisor
	MCU	Multiplexer Control Unit
	MDEP	Multinational Design Evaluation Programme
	MDFP	Motor Driven Fire Pump
	MDMA	Multi Attribute Decision Analysis
	MD-RFP	Motor Driven Reactor Feedwater Pump
	MDSL	Master Document Submission List
	ME	Mechanical Engineering
	ME	Medium Efficiency
	MEB	NRC Mechanical Engineering Branch
	MEH	Mechanical, Electrical and HVAC
	MET	Meteorological Observation System
	MF	Multiple Failure
	MFBE	Misloaded / Mislocated Fuel Bundle Event
	MFLE	Mislocated Fuel Loading Error
	MG	Motor-Generator
	MIC	Microscopic/Conversion
	MIL	United States Military Standard

MLHGR	Maximum Linear Heat Generation Rate
MMMF	Man Made Mineral Fibre
MOFB	Mis-Oriented Fuel Bundle
MOLF	Marine Offloading Facility
MOP	Main Turbine Main Oil Pump
MOP	Mechanical Over Power
MOV	Motor Operated Valve
MOX	Mixed Oxide
MP	Monitoring Post
MPC	Maximum Permissible Concentration
MPC	Multi-Purpose Containers
MPCWLL	Maximum Primary Containment Water Level Limit
MPS	Missing Pellet Surface
MRBM	Multi-Channel Rod Block Monitor
MS	Main Steam System
MSC	Miscellaneous Non-Radioactive Drain Transfer System
MSF	Main Steam Flow
MSIV	Main Steam Isolation Valve
MSIVC	Main Steam Isolation Valve Closure
MSL	Main Steamline
MSL	Mean Sea Level
MSL	Master Submission List
MSL	Main Steam Lines
MSLB	Main Steam Line Break
MSLBA	Main Steam Line Break Accident
MSQA	Management of System and Quality Assurance
MSR	Moisture Separator Reheater
MSR-MDT	MSR Moisture Separator Drain Tank
MSR-1DT	MSR 1st Stage Reheater Drain Tank
MSR-2DT	MSR 2nd Stage Reheater Drain Tank
MSTR	Main Steam Tunnel Room
MSV	Mean Square Voltage
MSV	Main Stop Valve
MSW	Miscellaneous Solid Waste System
MTBF	Mean Time Between Failure
MTTR	Mean Time to Repair
MUWC	Make-up Water Condensate System
MUWCP	Make-up Water Condensate Pumps
MUWP	Make-up Water Purified System

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	MUX	Multiplexing System
	MV	Medium Voltage
	MVA	Million Volt Amps
	MVD	Medium Voltage Distribution System
	MVDS	Modular Vault Dry Storage
	MVP	Mechanical Vacuum Pump
	MW	Megawatt
	MWTC	Miscellaneous Waste Transfer Container
N	NATRASS	Nuclear Plant Advanced Transient Data Recording and Analysis Support System
	NB	Nuclear Boiler System
	NB	ASME Subsection NB
	NBS	Nuclear Business Standards (Hitachi-GE)
	NC	ASME Subsection NC
	NCLL	Normal Combustible Loading Limit
	NDA	Nuclear Decommissioning Authority
	NDE	Non-destructive Examination
	NDI	Non-destructive Inspection
	NDT	Non-Destructive Testing
	NDTT	Nil Ductility Transition Temperature
	NEA	Nuclear Energy Agency (of the OECD)
	NELS	Non-Class 1E Emergency Lighting Subsystem
	NEMA	National Electrical Manufacturers Association
	NF	Nuclear Fuel
	NFIS	New Fuel Inspection Stand
	NFV	New Fuel Vault
	NG	Nuclear Grade
	NI	Nuclear Island
	NIA	Nuclear Industry Association
	NISA	Nuclear and Industrial Safety Agency (JP)
	NMCA	Noble Metal Chemical Addition
	NMIS	Noble Metal Injection System
	NMS	Neutron Monitoring System
	NNLS	Non-Class 1E Normal Lighting Subsystem
	NOD	Non-Radioactive Oil Drain
	NPAR	Nuclear Plant Aging Research
	NPB	Non-Segregated Phase Bus
	NPC	Normal Packing Clearance
	NPP	Nuclear Power Plant

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	NPS	National Policy Statement
	NPS	Nuclear Power Station
	NPSH	Net Positive Suction Head
	NRA	Nuclea Regulation Authority (JP)
	NRC	Nuclear Regulatory Commission (US)
	NRD	Miscellaneous Non-Radioactive Drain System
	NRHX	Non-Regenerative Heat Exchanger
	NRW	Natural Resources Wales
	NSC	Nuclear Safety Committee
	NSCs	Nuclear Special Cranes
	NSD	Non-Radioactive Storm Drain
	NSEDP	Nuclear Safety and Environmental Design Principles
	NSEP	Nuclear Safety and Environmental Principles
	NSF	Nuclear Spent Fuel
	NSL	Nuclear Site License
	NSLS	Non-Class 1E Standby Lighting Subsystems
	NSOA	Nuclear Safety Operational Analysis
	NSS	Nuclear Safety Systems
	NSSD	Nuclear Safety Site Description
	NSSS	Nuclear Steam Supply System
	NSU	Neutron Source Unit
	NUCAMM	Nuclear Power Plant Control Complex with Advanced Man-Machine Interfaces
	NUREG	Nuclear Regulatory Commission Regulation (US)
	NW	Natural Water System
	NWC	Normal Water Chemistry
	NWL	Normal Water Level
	NZO	Natural Zinc Oxide
O	O&M	Operation and Maintenance
	OBE	Operating Basis Earthquake
	OCIS	Other C&I System
	ODYN	One Dimensional Dynamic Model
	OECD	Organisation for Economic Co-operation and Development
	OEF	Operational Experience and Feedback
	OG	Off-Gas System
	OGRA	Off-Gas System Rupture Accident
	OI	Oxygen Injection System
	OHSAS	Occupational health and safety management system
	OJEU	Official Journal of the European Union

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	OL	Operating License
	OLMCPR	Operating Limit Minimum Critical Power Ratio
	OLNC	On-Line NobleChem™
	OLTC	On-load Tap Charger
	OLU	Output Logic Unit
	ONR	Office for Nuclear Regulation (UK)
	ONR (CNS)	Civil Nuclear Security (part of the Office for Nuclear Regulation) (UK)
	OPEX	Operational Experience
	OR	Operating Rule
	ORE	Operator Radiation Exposure
	OSC	Operational Support Centre
	OSHA	Occupational Safety & Health Administration
	OSR	Operational Safety Report
	OSCGS	Outer Secondary Containment Grab Sampler
	OTPS	Over Temperature Protection System
	OTS	Operating Technical Specifications
P	P&D	Plumbing and Drainage System
	P&ID	Piping & Instrumentation Diagram
	P&ID	Process and Information Document for Generic Assessment of Candidate Nuclear Power Plant Design
	P/C	Power Centre
	P/E	Pneumatic-to-Electric Converter
	PAC	Plasma Arc Cutting
	PAC	Pre Application Consultation
	PADS	Tank and Equipment Pads
	PADT	Programming and Debugging Tool
	PAE	Project Application Engineering
	PAMS	Post Accident Monitoring System
	PAR	Passive Auto-catalytic Recombiner
	PASS	Post-Accident Sampling System
	PCB	Primary Containment Boundary
	PCC	Plant Capital Costs
	PCHS	Power Cycle Heat Sink
	PCI	Pellet Cladding Interaction
	PCI	Pre-Construction Information
	PCIS	Primary Containment Isolation System
	PCmSR	Pre-Commissioning Safety Report
	PCntlS	Plant Control System
	PCS	Process Control Systems

Appendix A: Abbreviations and Acronyms List

PCS	Plant Computer System
PCS	Power Conversion Systems
PCSR	Pre-Construction Safety Report
PCT	Peak Cladding Temperature
PCV	Primary Containment Vessel
PCV	Pressure Control Valve
PCVB	Primary Containment Vessel Boundary
PCW	Plant Chilled Water
PDC	Principal Design Criteria
PDDP	Pump Deck Differential Pressure
PDS	Plant Damage State
PED	Pressure Equipment Directive
PEP	Project Execution Plan
PFD	Probability of Failure per Demand
PFD	Process Flow Diagram
PFD	Plant Level Flat Display
PGA	Peak Ground Acceleration
PGC	Power Generation Costs
PHA	Peak Horizontal Acceleration
PHCS	Power Cycle Heat Sink
PIE	Postulated Initiating Event
PI/O	Process Input/Output
PIP	Plant Investment Protection
PLC	Programmable Logic Conrtroller
PLR	Part-Length Rod
PM	Preventative Maintenance
PMF	Probable Maximum Flood
PML	Principia Mechanica Limited
POCO	Post Operations Clean-Out
POS	Plant Operating States
POSR	Pre-Operational Safety Report
PP	Physical Protection
PPE	Pre-Project Engineering
PPE	Personal Protective Equipment
PRA	Probabilistic Risk Assessment
PRDF	Pressure Regulator Downscale Failure
PRM	Process Radiation Monitoring System
PRM	Power Range Monitor
PS	Pipe Space

Appendix A: Abbreviations and Acronyms List

	PS	Power Supply
	PSA	Probabilistic Safety Analysis
	PSA	Probabilistic Safety Assessment
	PSD	Power Spectral Density
	PSHA	Probabilistic Seismic Hazard Assessment
	PSI	Pre-Service Inspection
	PSR	Preliminary Safety Report
	PSRI	Pre-Service Regulatory Inspection
	PSSR	Pressure Systems Safety Regulations
	PT	Liquid Penetrant Test
	PUWER	Provision and use of Work Equipment Regulations
	PV	Pressure Vessel
	PVC	PolyVinyl Chloride
	PWR	Pressurized Water Reactor
	PWST	Purified Water Storage Tank
	PY	Per Year
Q	QA	Quality Assurance
	QAP	Quality Assurance Program
	QC	Quality Control
	QEDS	Qualified Examination Defect Size
	QMP	Quality Management Plan
	QMS	Quality Management System
R	R/B	Reactor Building
	R/W	Reactor Well
	R2P2	Reducing Risks Protecting People
	R&A	Requirements and Assumptions
	R&D	Research & Development
	RACC	Rod Action Control Cabinet
	RAW	Risk Achievement Worth
	RBC	Reactor Building Overhead Crane
	RBEEE	Reactor Building Emergency Electrical Equipment
	RBEEE/Z	Reactor Building Emergency Electrical Equipment Zone
	RC	Reactor Chemistry
	RC	Reinforced Concrete
	RC&IS	Rod Control & Information System
	RCA	Radiation Controlled Area
	RCC	Remote Communication Cabinet

Appendix A: Abbreviations and Acronyms List

RCCV	Reinforced Concrete Containment Vessel
RCIC	Reactor Core Isolation Cooling System
RCM	Reactor Coolant Makeup System
RCM	Reliability Centred Maintenance
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RCTM	Requirements Compliance Tracking Matrix
RCW	Reactor Building Cooling Water System
RCWY	Raceway System
RD	Radioactive Drain Transfer System
RDCF	Reactor Depressurization Control Facility
RECHAR	Recombiner and Ambient Temperature Charcoal Absorption
REPPIR	Radiation (Emergency Preparedness and Public Information) Regulations
REPPIR2001	Radiation (Emergency Preparedness and Public Information) Regulations 2001
RFC	Recirculation Flow Control System
RFI	Radio Frequency Interference
RFM	Refuelling Machine
RFP	Reactor Feedwater Pump
RFP-T	RFP Turbine
RG	Regulatory Guide (US)
RGP	Relevant Good Practice
RHR	Residual Heat Removal System
RHX	Regenerative Heat Exchanger
RI	Regulatory Issue
RIC	Reactor Island Complex
RIN	Reactor Internals
RIP	Reactor Internal Pump
RM	Recirculation Motor
RMC	Recirculation Motor Cooling System
RMHX	RIP Motor Heat Exchanger
RMISS	Recirculation Motor Inflatable Shaft Seal System
RMP	Recirculation Motor Purge System
RMU	Remote Multiplexing Unit
RO	Reverse Osmosis
RO	Regulatory Observation
ROA	Regulatory Observation Action
ROC	Republic of China
ROVA	Remotely Operated Valve Assembly
RP	Radiation Protection

	RP	Requesting Party
	RPA	Radiation Protection Adviser
	RPS	Reactor Protection System
	RPT	Recirculation Pump Trip
	RPV	Reactor Pressure Vessel
	RQ	Regulatory Query
	RRPS	Reference Rod Pull Sequence
	RRS	Reactor Recirculation System
	RSA	Radioactive Substances Activities
	RSM	Rod Server Module
	RSP	Remote Shutdown Panel
	RSR	Radioactive Substances Regulation
	RSS	Remote Shutdown System
	RSSR	Remote Shutdown System Panel Room
	RSTS	Remote Shutdown Transfer Switch
	RSW	Reactor Shield Wall
	RSW	Reactor Building Service Water System
	RT	Radiographic Test
	RTNDT	Reference Temperature For Nil Ductility Transition
	RUHS	Reserve Ultimate Heat Sink
	RVI	Reactor Vessel Instrument System
	RVSS	Reactor Vessel Support Structure
	Rw/B	Radwaste Building
	Rw/B MCR	Radwaste Building main Control Room
	RWA	Radioactive Waste Adviser
	RWE	Rod Withdrawal Error
	RWM	Rod Worth Minimizer
	RWM	Radioactive Waste Management Limited
	RWP	Radiation Work Permit
S	S&PC	Steam and Power Conversion
	S&W	Stone and Webster Engineering Company
	S/B	Service Building
	S/C	Suppression Chamber
	S/P	Suppression Pool
	SA	Station Service Air System
	SA	Severe Accident
	SAA	Severe Accident Analysis
	SACF	Single Active Component Failure

Appendix A: Abbreviations and Acronyms List

SACS	Safety Auxiliary Control System
SAM	Sampling System
SAM	Process Sampling System
SAMDA	Severe Accident Mitigation Design Alternatives
SAMG	Severe Accident Management Guideline
SAP	Safety Assessment Principles
SAR	Safety Analysis Report
SAuxP	Safety Auxiliary Panel
SBO	Station Blackout
SBPC	Steam Bypass and Pressure Control System
SBWR	Simplified Boiling Water Reactor
SC	Shutdown Cooling
SCB	Secondary Containment Boundary
SCC	Stress Corrosion Cracking
SCDM	Safety Case Development Manual
SCDP	Safety Case Development Plan
SCF	Single Component Failure
SCG	Startup Coordinating Group
SCRAM	Reactor Emergency Shutdown (Safety Control Rod Insertion)
SCRRI	Selected Control Rod Run-In
SCSG	Safety Case Steering Group
SCV	Secondary Containment Vessel
SCWG	Safety Case Working Group
SD	Smart Device
SDC	Safety Design Criteria
SDC	Shutdown Cooling
SDCP	Seal Drain Collector Pump
SDCT	Seal Drain Collection Tank
SDD	System Design Description
SDV	Screening Distance Value
SECY	Office of the Secretary of the Commission
SEP	Standby Electrical Power
SER	Safety Evaluation Report
SF	Spent Fuel
SFAIRP	So far As is Reasonably Practicable
SFC	Safety Functional Claim(s)
SFC	System Fault Condition
SFC	Single Failure Criterion
SFCHG	Spent Fuel Cask Handling Grapple

Appendix A: Abbreviations and Acronyms List

SFE	Spent Fuel Export
SFIS	Spent Fuel Interim Storage
SFISF	Spent Fuel Interim Storage Facility
SFP	Spent Fuel Storage Pool
SFP	Spent Fuel Pool
SFR	Safety Functional Requirement(s)
SFS	Spent Fuel Storage Facility
SGHWR	Steam Generating Heavy Water Reactor
SGTS	Standby Gas Treatment System
SHA	Seismic Hazard Assessment
SHE	Standard Hydrogen Electrode
SHEQ	Safety, Health, Environmental and Quality Management
SI	Structural Integrity
SIL	Safety Integrity Level
SIM	Training Simulator
SIT	Structural Integrity Test
SJAE	Steam Jet Air Ejector
SJR	Site Justification Report
SKI	Swedish Nuclear Power Inspectorate
SLA	Site License Application
SLC	Standby Liquid Control System
SLC	Site License Company
SLD	Standby Liquid Drain
SLG	Site Licence Grant
SLMCPR	Safety Limit Minimum Critical Power Ratio
SLU	Safety Logic Unit
SMA	Seismic Margins Analysis
SMA	Seismic Margin Assessment
SME	Subject Matter Expert
SMP	Software Management Plan
SMS	Seismic Monitoring System
SoDA	Statement of Design Acceptability
SOE	Single Operator Error
SOL	Solidifying System
SOP	Severe Accident Operation
SOP	System Operating Procedure
SOT	System Operational Transients
SPC	Suppression Pool Cooling
SPC	Safety Properties Claim(s)

Appendix A: Abbreviations and Acronyms List

SPCU	Suppression Pool Clean-up System
SPD	Suppression Pool Water Drainage System
SPDS	Safety Parameter Display System
SPL	Set Point List
SPT	Suppression Pool Water Surge Tank
SPTM	Suppression Pool Temperature Monitoring System
SQEP	Suitably Qualified and Experienced Personnel
SR	Surveillance Requirements
SREE	Safety-Related Electrical Equipment
SRI	Selected Rod Insertion
SRM	Source Range Monitor
SRMS	Solid Radwaste Management System
SRNM	Start-up Range Neutron Monitor
SROA	Safety-Related Operator Action
SRP	Standard Review Plan (US)
SRV	Safety Relief Valve
SRVN	Safety Relief Valve Nitrogen Gas Supply System
SRW	Solid Radwaste System
SS	Stainless Steel
SS	Spent Sludge System
SS	Site Specific
SSA	Strategic Siting Assessment
SSAR	(The ABWR) Standard Safety Analysis Report (US)
SSCs	Systems, Structures and Components
SSE	Safety, Security and Environment
SSI	Soil-Structure Interaction
SSLC	Safety System Logic and Control
SSLS	Class 1E Associated Standby Lighting Subsystem
SSPC	Steel Structures Painting Council
ssPCSR	site specific Pre-Construction Safety Report
SSSI	Site of Special Scientific Interest
STC	Surveillance Test Controller
STPT	Simulated Thermal Power Trip
STR/AP	Scram Time Test Recording/Analysis Panel
STS	Sewage Treatment System
STTP	Scram Time Test Panel
SUMIT	Spectral Unit Module Initial
SUS	Secondary Unit Substation
SW	Switch

	SW	Sea Water
	SWC	Surge Withstand Capability
	SWC	Generator Stator Winding Cooling System
	SWL	Safe Working Load
	SWMS	Solid Waste Management System
	SWSA	Solid Waste Storage Area
	SWSD	Service Water Storm Drain
	SWTC	Standard Waste Transport Container
	SWYD	Switchyard
T	T&M	Test and Maintenance
	T/B	Turbine Building
	T/D-RFP	Turbine Driven Reactor Feedwater Pump
	TAF	Top of Active Fuel
	TAG	Technical Assessment Guide
	TAGSI	Technical Advisory Group on Structural Integrity
	TBC	To Be Confirmed
	TBD	A defined Plant Damage State for PSA
	TBP	Turbine Bypass System
	TBU	A defined Plant Damage State for PSA
	TBV	Turbine Bypass Valve
	TC	Training Centre
	TC	A defined Plant Damage State for PSA
	TCF	Total Core Flow
	TCOM	Transmission Commission
	TCOM	Transmission Communication
	TCPA	Town and Country Planning Act 1990
	TCS	Turbine Control System
	TCV	Temperature Control Valve
	TCW	Turbine Building Cooling Water System
	TCW-HEX	Turbine Building Cooling Water Heat Exchanger
	TCWP	Turbine Building Cooling Water Pump
	TD	Tornado Damper
	TDH	Total Developed Head
	TD-RFP	Turbine Driven Reactor Feedwater Pump
	TE	Temperature Element
	TEDE	Total Effective Dose Equivalent
	TEMA	Tubular Exchanger Manufacturers Association
	TEPCO	Tokyo Electric Power Company, Inc.

	TG	Top Guide
	TG	Technical Governance
	T-G	Turbine Generator
	TGS	Turbine Gland Steam System
	TGSCC	Transgranular Stress Corrosion Cracking
	THA	Time-History Accelerographs
	THERP	Technique for Human Error Rate Prediction
	THISO	Third Height International Standards Organisation
	TI	Turbine Island
	TIP	Traversing In-Core Probe
	TIU	Technician Interface Unit
	TLU	Trip Logic Unit
	TMI	Three Mile Island
	TN	Transmission Network
	TOC	Table of Contents
	TOC	Top of Core
	TOC	Total Organic Carbon
	TOP	Thermal Over Power
	TOR	Tolerability of Risks From Nuclear Power Stations
	TPRD	Turbine Plant Radioactive Drain System
	TQUV	A defined Plant Damage State for PSA
	TQUX	A defined Plant Damage State for PSA
	TR	Topic Report
	TRS	Test Response Spectra
	TSC	Technical Support Centre
	TSC	Technical Support Contractor
	TSO	Transformer System Operator
	TSV	Turbine Stop Valve
	TSW	Turbine Building Service Water System
	TSWP	Turbine Building Service Water Pump
	TV	Tank Vent Treatment System
	TVAPS	Time Varying Axial Power Shape
	TW	A defined Plant Damage State for PSA
U	U/D	Upper Drywell
	UHP	Ultra High Pressure
	UHRS	Uniform Hazard Response Spectra
	UHS	Ultimate Heat Sink
	UHS	Uniform Hazard Spectra

Appendix A: Abbreviations and Acronyms List

	UK	United Kingdom
	UKCIP02	UK Climate Impact Programme 2002
	UKCP09	UK Climate Projection 2009
	UL	Underwriters Laboratory
	UOP	Unit Operating Procedure
	UPS	Uninterruptible (A.C.)Power Supply
	URD	Utility Requirements Document
	URS	Ultimate Rupture Strength
	US NRC	U.S. Nuclear Regulatory Commission
	USE	Upper Shelf Energy
	USMA	Uniform Support Motion Response Spectrum Analysis
	UT	Ultrasonic Test
	UT	Unit Transformer
V	V&V	Verification and Validation
	V/H	Vertical to horizontal
	VAC	Vital AC Power Supply
	VAC	Volts Alternating Current
	VB	Vacuum Breaker
	VBWR	Vallecitos Boiling Water Reactor
	VDC	Volts Direct Current
	VDU	Visual Display Unit
	VGL	Valve Gland Leakage Treatment System
	VGS	Valve Gland Seal Water System
	VHI	Very High Integrity
	VHIC	Very High Integrity Component
	VHL	Very Heavy Lift
	VLC	Vent Line Clearing
	VLLW	Very Low Level Waste
	VMS	Revolving Solid Vibration Monitoring System
	VWO	Valves-Wide-Open
W	W/W	Wetwell
	WAC	Waste Acceptance Criteria
	WAMH	Waste Addition and Mixing Head
	WANO	World Association of Nuclear Operators
	WBC	Whole Body Counter
	WDP	Wide Display Panel
	WENRA	Western European Nuclear Regulators Association

Appendix A: Abbreviations and Acronyms List

WJP	Water Jet Peening
WRC	Welding Research Council
WRL	Wide Range Level
WST	Water Storage Tank
ZNIS	Zinc Injection System
ZSI	Zone Selective Interlocks