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PSR Part B Chapter 15 BDBA, Severe Accidents Analysis and Emergency Preparedness

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15.1 INTRODUCTION

The Fundamental Purpose of the Generic Design Assessment (GDA) Safety, Security and Environment Case (SSEC) is to demonstrate that the generic Small Modular Reactor (SMR)-300 can be constructed, commissioned, operated, and decommissioned on a generic site in the United Kingdom (UK) to fulfil the future licensee's legal duties to be safe, secure and protect people and the environment, as defined in Part A Chapter 1 Introduction [1] .

The Fundamental Purpose is achieved through the Fundamental Objective of the Preliminary Safety Report (PSR), which is to summarise the safety standards and criteria, safety management and organisation, claims, arguments and intended evidence to demonstrate that the generic SMR-300 design risks to people are likely to be tolerable and As Low as Reasonably Practicable (ALARP) [1].

Part B Chapter 15 of the PSR presents the Claims, Arguments and intended Evidence (CAE) for the Beyond Design Basis Accidents (BDBA), Severe Accident Analysis (SAA) and Emergency Preparedness (EP) topic that underpins the design of the Generic SMR-300.

15.1.1 Purpose and Scope

The Overarching SSEC Claims are presented in Part A Chapter 3 'Claims, Arguments & Evidence' [2].

This chapter (Part B Chapter 15) links to the overarching claim through Claim 2.1:

Claim 2.1: The nuclear safety assessment identifies plant initiating events and specifies the requirements for safety measures such that safety functions are fulfilled, informs operational and emergency arrangements and demonstrates that risk is tolerable and As Low As Reasonably Practicable (ALARP).

As set out in Part A Chapter 3 [2], Claim 2.1 is further decomposed across several disciplines which are responsible for development of supporting nuclear safety assessments. This chapter's purpose is to demonstrate that there is a robust methodology for the identification and assessment of fault conditions beyond the design basis relevant to the Holtec SMR-300 design. This will be done by satisfying Claim 2.1.3.

Claim 2.1.3: Beyond design basis faults and severe accidents are appropriately identified and risk assessed to be tolerable and As Low As Reasonably Practicable (ALARP).

Further discussion on how the Level 3 claim is broken down into Level 4 claims and how the Level 4 claims are met is provided in sub-chapter 15.4.

The scope of this chapter covers the matter related to Beyond Design Basis, Severe Accidents (SA) and Emergency Arrangements (EA) as set out in sub-chapter 15.2.

Sub-chapter 15.5.1 covers the codes and standards associated with the topics.

Sub-chapter 15.3 covers identified additional reports which complement Chapter 15 in the GDA process.

Further sub-chapters (15.6, 15.7, 15.8 and 15.9) focus on the identified claims and their arguments as well as the interpretation of the safety objectives in the context of SMR-300.

Finally, sub-chapter 15.10 provides a technical summary of how the claims for this Chapter have been achieved, together with a summary of key contributions from this chapter to the overall ALARP. Sub-chapter 15.10 also discusses any GDA commitments that have arisen.

Excluded from the Part B Chapter 15 scope are design analysis and safety analysis which are dedicated to Severe Accidents Management and those resulting from Level 2 Probabilistic Safety Assessment (PSA) as SMR-300 PSA is still under development. Future iterations of this chapter will discuss such analysis when supporting documentation has matured. Such information and its scope are discussed in [3] and [4].

A master list of definitions and abbreviations relevant to all PSR Chapters can be found in Part A Chapter 2 General Design Aspects and Site Characteristics [5].

15.1.2 Assumptions

Assumptions which relate to this topic have been formally captured in the Commitments, Assumptions and Requirements process [6]. Further details of this process are provided in Part A Chapter 4 [7].

15.1.3 Interfaces with other SSEC Chapters

The Beyond Design Basis Accidents, Severe Accident Analysis and Emergency Preparedness chapter interfaces with the following PSR chapters.

Part A Chapter 2 General Design Aspects and Site Characteristics [5]: The chapter presents an overview of the generic plant description, including the main buildings and structures and their associated systems, these provide inputs to the safety evaluation for Design Extension Conditions (DEC-A and DEC-B) in Chapter B15.

Part A Chapter 5 Summary of ALARP [8]: The chapter presents the ALARP methodology and ALARP justifications for the SMR-300, which summarises the ALARP assessment of BDBA and Severe Accidents (SA).

Part B Chapter 1 Reactor Coolant System and Engineered Safety Features [9] and Part B Chapter 2 Reactor Fuel and Core [10] provide the substantiation of the Reactor Coolant System (RCS) and of the Safety Systems which are taken into consideration for the DECs analysis and the generic nuclear data for Beyond Design Basis (BDB) and SAA.

Part B Chapter 10 Radiological Protection [11] for derivation of the operational source term noting that Chapter B15 should provide the BDB and Severe Accidents (SA) source terms.

Part B Chapter 14 Design Basis Analysis (DBA) [12]: Chapter B15 extends the Design Basis Analysis of Chapter B14, which provides initiating events for sequences identification and methodology for hazard identification.

Part B Chapter 16 Probabilistic Safety Assessment (PSA) [13]: Chapter B16 provides PSA results to support the identification of BDB and SA events and shows that the total risks and exposure of public and workers from SA events can meet specified radiation protection targets. Chapter B15 provides the thermal-hydraulic analysis results and source term input to PSA.

Part B Chapter 17 Human Factors (HF) [14]: Chapter B17 substantiates the claims on operator actions under BDB and SA conditions.

Part B Chapter 18 Structural Integrity (SI) [15]: Chapter B18 interfaces with Chapter B15 to assess extreme scenarios and their effects on SI and enhance structural robustness against SAs.

Part B Chapter 20 Civil Engineering [16]: Chapter B20 presents the design substantiation of civil structures. Chapter B15 may provide thermal-hydraulic conditions in internal containment under SA for design of civil structures.

Part B Chapter 23 Reactor Chemistry [17]: Chapter B23 provides accident processes for BDB and SA events, and source term of fission product and combustible gases to support the understanding of accident chemistry.

15.2 OVERVIEW OF BEYOND DESIGN BASIS, SEVERE ACCIDENTS AND EMERGENCY PREPAREDNESS

In conjunction with PSR Part B Chapter 14 [12] and PSR Part B Chapter 16 [13], this chapter provides a description of the safety analyses performed to assess the safety of the plant in normal operation and in response to postulated initiating events and accident scenarios on the basis of established acceptance criteria.

The scope of faults to be considered in the SMR-300 safety assessment is illustrated in Table 1 which shows the frequency range of design basis accidents and those of BDBA and Severe Accidents.

Table 1: Plant Condition Grouping

Plant Condition Class	Design Basis Condition Class	Initiating Event Frequency (IEF) Range (y^{-1})
Normal operation	DBC1	IEF>1
Anticipated operational occurrences	DBC2	1>IEF>1E-02
Design basis accidents	DBC3a	1E-02>IEF>1E-03
	DBC3b	1E-03>IEF>1E-04
	DBC4	1E-04>IEF>1E-05
Design Extension Conditions (DECs) with or without significant core disruption – beyond design basis or SAs	DEC – A (without core damage) DEC-B (progressing to core damage)	IEF>1E-05
Accident with releases requiring implementation of emergency countermeasures	Off-site emergency, severe accidents	IEF>1E-05

It aims to demonstrate the plant is capable of preventing, controlling and mitigating sequences which are outside of the design basis. In addition, such information is further used to inform the Accident Management (AM) approach and activities.

The Structures, Systems, and Components (SSCs) playing roles within the scope of this Chapter are outlined in sub-chapter 15.8.

The aim of PSR Part B Chapter 15 is to address the evaluation of the Design Extension Conditions (DECs) for the Generic SMR-300 and to demonstrate that accidents that have the potential to lead to severe consequences have been systematically analysed, and the analysis is used to identify appropriate preventative and mitigating measures beyond those derived from the DBAA and provided in PSR Part B Chapter 14 [12].

15.2.1 BDBA and DEC

UK Regulatory expectations on BDBA are summarised in the ONR-GDA-007 New Nuclear Power Plants: Generic Design Assessment Technical Guidance [18]. In this context, the BDBA are recognised as being incorporated within DEC conditions. Here, the two nomenclatures can be considered synonymous. The following scenarios should be considered in the BDBA analysis:

- Initiating events that could lead to situations beyond the capability of safety systems that are designed for design basis accidents.
- Frequent design basis accidents combined with multiple failures that prevent the safety systems from performing their intended function to control the postulated initiating event.
- Credible postulated initiating events involving multiple failures causing the loss of a safety system while this system is used to fulfil its function as part of normal operation.

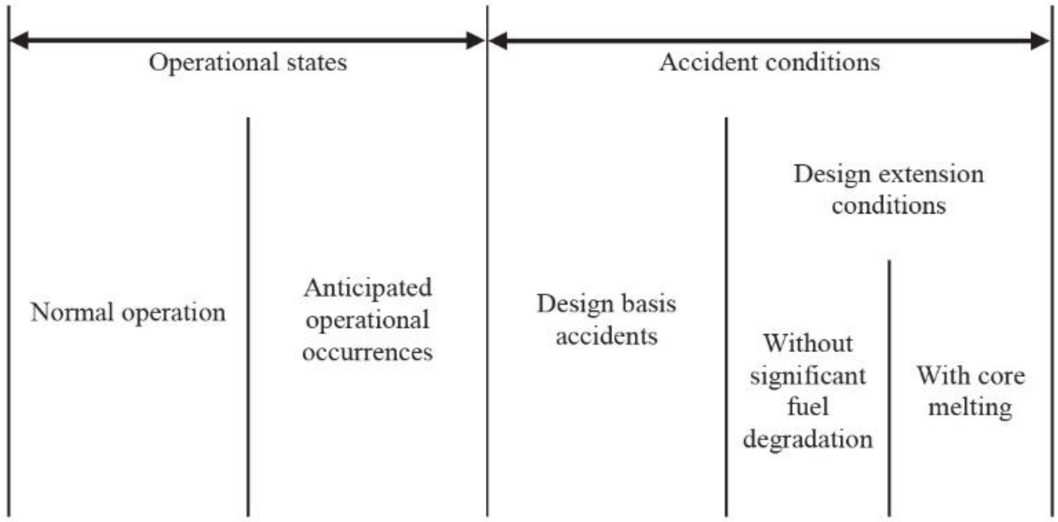

Figure 1: Plant States Considered in Design for a Nuclear Power Plant

As the United States (US) has been at the forefront of the development of light water reactor technology for over 70 years, in particular the Pressurised Water Reactor (PWR), it is considered that the regulatory arrangements and requirements set out by the US Nuclear Regulatory Commission (NRC) represent international good practice. However, this analysis will be organised and expanded where appropriate to adequately consider UK Relevant Good Practice (RGP) and the relevant Office for Nuclear Regulation (ONR) Safety Assessment Principles (SAPs) [19] and Technical Assessment Guide (TAGs), 'Codes and Standards'.

The DBAA of the Generic SMR-300 is addressed in Part B Chapter 14 [12]. The ONR SAPs [19] states (paragraph 628) that when initiating faults are excluded from the DBAA, the safety case should still demonstrate that the resultant risks are ALARP.

DEC refer to events of low frequency where the conditions may be more onerous than those identified in the DBAA as well as events involving combination of initiators and leading to high consequences (Figure 1). In the UK context, when the potential consequences of these Beyond Design Basis Events (BDBEs) is more severe, ONR refers to these as 'Severe Accidents' (SA). SAs are well defined in the SAPs [19] (paragraph 664) in terms of radiological consequences and societal risk. As stated in the TAG NS-TAST-GD-007 guide on SAA [20], the SA is often associated with significant core degradation (DEC-B). In the international context, following International Atomic Energy Agency (IAEA) and Western European Nuclear

Regulators' Association (WENRA) approaches, SAs are classified as DEC-B, namely DEC associated with significant core damage. DEC-A, instead, are the ones associated to DEC occurring without significant fuel degradation. This last category can be identified with the 'Beyond Design Basis Events' (BDBEs) as they are defined in ONR-GDA-007 guide [18].

15.2.2 Accident Management and Emergency Preparedness

Accident Management (AM) is expected to prevent the escalation of the event to a SA, to mitigate the consequences of the accident, and to ultimately achieve a long-term stable condition for the plant. Long-term safe stable state is defined as the termination of core damage if it has begun, maintaining containment where possible, minimising radiological release on and off site and the plant being returned to a controlled state. SAs are initially managed by the facility operators using Emergency Operating Procedures (EOPs) and then transition to Severe Accident Management Guidelines (SAMGs) if core damage cannot be prevented.

Emergency Preparedness (EP) is defined in the IAEA Safety Glossary [21] as "The capability to take actions that will effectively mitigate the consequences of an emergency for human health and safety, quality of life, property, and the environment". EP represents the fifth and final level of Defence in Depth (DiD) applied in the design of the SMR-300. EP is primarily established to prepare for a radiation emergency and mitigate the consequences in case of an occurrence by taking all reasonably practicable measures, as required by UK legislation.

15.2.3 SMR-300 Severe Accident Philosophy

The approach to DiD for the SMR-300 is set out in HI-2240251, Holtec SMR Top-Level Plant Design Document [22] and described in PSR Chapter A2. DiD helps to establish that the three basic safety functions (controlling the power, cooling the fuel, and confining the radioactive material) are preserved, and that radioactive materials do not reach people or the environment.

The SMR-300 applies a comprehensive (DiD) strategy to ensure the highest level of nuclear safety and resilience. This framework includes five distinct, reinforcing levels, each designed to be independently robust, highly reliable, and effective under a wide range of operating and accident conditions. The goal of each level of protection and the essential means of achieving them are shown in Table 2.

Table 2: Holtec SMR Defence-in-Depth Philosophy

Level	Goal	Essential Means
Level 1	Prevention of abnormal operation and failures	Conservative design and high quality in construction and operation
Level 2	Control of abnormal operation and detection of failures	Control, limiting and protection systems and other surveillance features
Level 3	Control of accidents within the design basis	ESF and accident procedures
Level 4	Control of severe plant conditions, including prevention of accident progression and mitigation of the consequences of severe accidents	Complementary measures and accident management
Level 5	Mitigation of radiological consequences of significant releases of radioactive materials	Off-site emergency response

The philosophy for the management of severe accidents has been developed based on the following approach:

a) Understanding Severe Accident Phenomena.
b) Protection against BDBE
c) Defining the Safety Features provided to manage Severe Accidents.
d) Defining the Emergency Operating Procedures (EOPs) and Severe Accident Management Guidelines (SAMGs).

15.2.3.1 Severe Accident Phenomena

One of the main objectives of the SMR-300 SAA is to identify all potential severe accident phenomena that should be considered, as well as the phenomena that can be excluded from analyses. For each phenomenon that is required to be prevented or mitigated, a severe accident management strategy and appropriate safety features should be identified. This includes High Pressure Melt Ejection (HPME) and direct containment heating which has the potential to cause a significant challenge to containment structures and lead to a large radioactive release.

15.2.3.2 Protection against BDBE

BDBE may lead to severe accidents and core damage. Hazards like flooding, seismic events and extended loss of power are explicitly addressed in plant layout, system survivability and emergency preparedness. Critical systems are seismically qualified and physically separated to ensure at least one train survives any postulated initiating event.

15.2.3.3 Safety Features provided to manage Severe Accidents

The SMR-300 is designed with multiple diverse systems to manage the development and consequences of severe accidents. These systems include:

15.2.3.3.1 Passive Core Cooling System

The Passive Core Cooling (PCC) system is an Engineered Safety Feature providing passive decay heat removal in response to events which result in the loss of normal cooling capability. There is no reliance on non-safety active systems, as protection is provided via passive means. The PCC comprises:

- Primary Decay Heat Removal System (PDH).
- Secondary Decay Heat Removal System (SDH).
- Automatic Depressurisation System (ADS).
- Passive Core Makeup Water System (PCM).

The PDH and SDH provide primary protection in the form of emergency decay heat removal when the normal heat rejection path is lost. Each decay heat removal system is capable of removing core decay heat and bringing the plant to a safe shutdown condition.

The ADS and PCM provide primary protection in the form of controlled depressurisation and safety injection of highly borated water, respectively, for DBAs resulting in loss of reactor coolant. The PCM and ADS will ensure sufficient RCS inventory to prevent fuel failure during the event and remove decay heat for 72 hours without operator intervention.

15.2.3.3.2 Passive Containment Heat Removal System (PCH)

The PCH maintains the Containment Structure (CS) atmospheric pressure and temperature within design limits in the event of a postulated accident by utilising the metal CS and the water inventory in the Annular Reservoir (AR). The PCH is a completely passive system that removes heat from the containment atmosphere. It does not require any actuations or an actuation signal to perform its safety functions.

15.2.3.3.3 In-Vessel Retention (IVR)

IVR is achieved through the successful depressurisation of the RCS combined with external cooling of the RPV. Depressurisation of the RCS is achieved using ADS-1 or ADS-2 which are reliable and diverse from one another to achieve depressurisation. Ex-vessel cooling is a passive system, with optional and diverse active operations, to manage molten core material (corium), preventing reactor vessel breach or containment basemat melt-through. This is achieved via a transfer of water from the PCM water tank to the reactor cavity. Hydrogen Management is undertaken using passive catalytic recombiners or igniters to prevent hydrogen accumulation and explosion post-core damage.

15.2.3.3.4 Post Accident Monitoring (PAM) System

Post Accident Monitoring is provided by displays of parameters as part of the Plant Safety System (PSS) and Plant Control System (PCS). The PSS presents Type B&C variables, the PCS Type D, E, F variables. The US / IAEA approach to Post Accident Monitoring (PAM) differs from the approach taken in the UK. There is no separate dedicated PAM Instrumentation and Control system and further work is required post GDA to explore the options for the system (C_C&I_083).

15.2.3.3.5 Environmentally Qualified Sensors and Control Systems

Environmentally qualified sensors and control systems that remain functional during severe accidents are provided in the design (e.g., wide-range temperature, pressure, and radiation sensors). Further information can be found in HI-2231065, Decision Paper on Environmental Qualification of Components Inside Containment [23].

15.2.3.3.6 Emergency Response Facilities (ERFs)

Dedicated Emergency Response Facilities (ERFs) with features, equipment and simulation tools to support decision-making under stress are provided in the design.

15.2.3.4 Severe Accident Management Guidelines (SAMGs)

Severe Accidents are initially managed by the facility operators using Emergency Operating Procedures (EOPs) and then transition to Severe Accident Management Guidelines (SAMGs) if core damage cannot be prevented.

In the extremely unlikely scenario of a severe accident where core melt is imminent or occurring, the SMR-300 safety strategy shifts to manage the accident's progression and mitigate its consequences. In BDBEs involving core melt, the SMR-300 no longer relies solely on PDH, SDH and the ADS Stages 1 and 2, particularly if those systems have failed, are unavailable, or the accident has progressed beyond their effective window. In these cases, alternative measures incorporated may be available to achieve reactor depressurisation and facilitate core cooling, primarily as part of SAMGs. These include, but are not limited to, using diverse safety and non-safety related systems like the Chemical and Control system letdown capability, RPV head vent valves, PDH vents, etc. The list of available accident management strategies and means to implement them will be identified and reasonable assurance that the equipment will survive to perform its function within the severe environment will be provided during development of SAMG framework.

The SAMGs include plant-specific guidelines for operators and emergency teams to respond to core damage, containment threats and potential releases. Operators and emergency teams undertake comprehensive simulator training in order to practice the EOPs and SAMGs.

15.3 TOPIC BASIS AND SUPPORTING DELIVERABLES

To support the GDA process and address differences between US and UK regulations, several additional reports are issued which set the expectations and the strategy for the BDBA, SA and EP topic as part of Chapter 15. These additional reports serve as a cohesive link for these topics, defining their scope and describing the planned next steps in safety demonstration in the UK regulatory context.

15.3.1 Safety Concept for Severe Accidents Report

The safety concept report [3] has been developed to summarise and describe the main principles and essential features of the safety concept used in SMR-300 nuclear power plant design and to provide information concerning the overall plant design in terms of safety. The document describes the structured safety approach aiming to ensure that the fundamental safety functions can be maintained without interruption throughout the operating life of the plant. The report serves the Beyond Design Basis Accidents and Severe Accident topic addressed in this Chapter, although it covers areas in the Design Basis and provides links to some of the Engineered Safety Features and systems in the design. It addresses the faults defined as Design Extension Conditions and is intended to support the development of the Holtec's SMR-300 GDA PSR and future detailed analysis.

A detailed list of plant systems and general engineered safety features is given in the respective PSR chapters, design descriptions and supporting documents [3], [9], [24], [25]. Actual initiation in the accident control and mitigation process is subject to further analysis and implementation in the Emergency Operating Procedures and Accident Management Guidelines.

15.3.2 Accident Management Program Report

The Accident Management Program report [4] is a key document supporting the design and its safety justification with respect to BDBA and SA. The report outlines the key elements that the team responsible for preparing, developing, and implementing a plant-specific Accident Management Program (AMP) at a nuclear power plant. It provides guidance and a basis for further AM development, up to the stage where site-specific conditions have impact on the further event progression. The AM report focuses on the fourth level of defence-in-depth, including transitions from the third level and into the fifth level. It follows best practices in accident management and emergency preparedness development and represents an integral part of the ALARP demonstration.

15.3.3 ALARP Demonstration of Severe Accidents and Emergency Preparedness Report

The SMR-300 ALARP Demonstration of Severe Accidents and Emergency Preparedness report [26] is part of the additional deliverables supporting Holtec's SMR-300 GDA for the Severe Accidents and Emergency Preparedness topic. It presents the overall approach for the demonstration of the principle of As Low As Reasonably Practicable (ALARP) for the Beyond Design Basis Accident and SA. The report outlines the key aspects of the ALARP process, including the sources of information, and how it aligns with the overall safety strategy. As part of the safety case, the report contributes to the arguments in support of the claim that the generic SMR-300 can be constructed, operated, and decommissioned on a generic site in the UK to fulfil the future licensee's legal duties to be safe, secure and protect people and the environment.

Together with the other additional deliverables/reports in support of the severe accident topic, the report contributes to the overall safety representation and provides a BDBA-specific approach to safety demonstration.

15.4 BEYOND DESIGN BASIS, SEVERE ACCIDENTS AND EMERGENCY PREPAREDNESS CLAIMS, ARGUMENTS AND EVIDENCE

This chapter presents the BDBA, SA and EP aspects for the generic SMR-300 and therefore directly supports Claim 2.1.3.

Claim 2.1.3: Beyond design basis faults and severe accidents are appropriately identified and risk assessed to be tolerable and As Low As Reasonably Practicable (ALARP).

The Fundamental Purpose follows a golden thread throughout the SSEC to CAE via the objectives of the PSR, the Preliminary Environmental Report (PER) and the Generic Security Report (GSR). The overarching SSEC claims and the philosophy for their architecture is presented in PSR Part A Chapter 3 [2].

This chapter contributes directly to Claim 2.1, which is focused on the demonstration of the design and that the SSCs that form the design, are developed to ensure they meet the relevant safety requirements and appropriate codes and standards.

This Chapter is predominantly focused around three main areas:

- Deriving safety functional requirements to be delivered by SSCs to manage severe accidents.
- Analysis of severe accidents to demonstrate that the identified safety features ensure relevant safety objectives and targets are met.
- The approach adopted for developing accident management and emergency preparedness arrangements.

Claim 2.1.3 has been further decomposed into five further sub-claims within PSR Part B Chapter 15 to provide confidence that the requirements of BDBA, SAA and EP are met.

Claim 2.1.3.1 presents the codes and standards related to BDBA, SAA and AM and EP used in the US and UK, together with applicable regulations and guidance.

Claim 2.1.3.2 and Claim 2.1.3.3 present the analysis (for DEC-A and DEC-B respectively) to demonstrate that the plant can reach a long-term safe state following a severe accident.

Claim 2.1.3.4 supports the derivation of safety features by identifying and ensuring SSCs are correctly specified in terms of safety functions and classification, derived from the safety analysis.

Claim 2.1.3.5 covers accident management and emergency preparedness.

Table 3 shows the breakdown of Claim 2.1.3, identifies which chapter of this PSR these claims are demonstrated to be met to a maturity appropriate for PSR v1.

Table 3: Claims Covered by Chapter B15

Claim No	Claim	Chapter Section
2.1.3.1	Beyond design basis and severe accidents are characterised and evaluated using appropriate methodologies taking due cognisance of RGP and OPEX.	15.5 Beyond Design Basis, Severe Accidents and Emergency Preparedness Codes and Standards
2.1.3.2	Deterministic analysis of DEC-A events (beyond design basis accidents not resulting in core damage) confirms the absence of “cliff edge” effects and demonstrates that the facility can be brought into a long term safe, stable state with maintained containment functions.	15.6 Deterministic analysis of DEC-A events
2.1.3.3	Severe accident analysis of DEC-B events (beyond design basis accidents with core damage) demonstrates that the facility can be brought into a long term safe, stable state with maintained containment functions.	15.7 Demonstration of plant response to DEC-B events
2.1.3.4	Additional reasonably practicable safety functions and safety measures are identified, categorised and classified based on their importance to nuclear safety for the purposes of Severe Accident management.	15.8 Safety means for Accident control and mitigation
2.1.3.5	Accident management and emergency preparedness take all reasonably practicable measures to prepare for possible accidents, and to mitigate their consequences should they occur.	15.9 Accident management and emergency preparedness

Appendix A provides a full Claims, Arguments and Evidence mapping for Chapter B15, which includes any lower-level claims, arguments and evidence needed to support the Claims in the table above. This includes identification evidence available at PSR v1 and aspects for future development of evidence to support these claims beyond PSR v1.

15.5 BEYOND DESIGN BASIS, SEVERE ACCIDENTS AND EMERGENCY PREPAREDNESS CODES AND STANDARDS

Claim 2.1.3.1: Beyond design basis and severe accidents are characterised and evaluated using appropriate methodologies taking due cognisance of RGP and OPEX.

This Sub-chapter justifies the code and standards related to BDBA, SAA and AM and EP used in the US and UK, together with applicable regulations and guidance. These codes and standards outline appropriate methodologies to be used considering RGP and OPEX.

15.5.1 Codes, Standards and Methodologies used for the BDBA and SAA of the SMR-300

The primary US guidance and requirements used in the development of the SMR-300 is provided by the following.

- 10 CFR 50.155 Mitigation of beyond-design-basis events [27].

Strategies and guidelines to mitigate BDB external events from natural phenomena are developed assuming a loss of all ac power concurrent with either a loss of normal access to the ultimate heat sink or, for passive reactor designs, a loss of normal access to the normal heat sink. This applies to both reactor and Spent Fuel Pool (SFP) and aligns with international best practice.

- HPP-160-3018, Design Standard for Severe Accident Design and Analysis Strategy [28].

The purpose of this Design Standard is to provide a systematic framework to address SAs in the design based on the regulatory requirements specified in Canadian Nuclear Safety Commission (CNSC) and US NRC regulatory documents. It assembles all relevant requirements and identified RGP associated with CNSC and US NRC regulatory documents. Additional documentation, including IAEA documents that contain useful information pertinent to SAs design are also identified as guidance to designers and safety analysts.

The Design Standard has been reviewed for applicability to SMR-300 with regards to codes and standards identified and has been deemed appropriate.

The other US guidance related to these topics is specified in Table 4.

Table 4: US Guidance

Label	Title	Revisions
NRC RG 1.206	Combined License Applications for Nuclear Power Plants (PWR Edition) [29]	June 2007
NUREG-0800, Chapter 19	Standard Review Plan – Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors [30]	Rev. 3, December 2015
NUREG-1555, Chapter 7	Standard Review Plans for Environmental Reviews for Nuclear Power Plants: Environmental Standard Review Plan [31]	August 2024
-	Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants [32]	August 1985
-	Policy Statement on Safety Goals for the Operations of Nuclear Power Plants [33]	August 1986

Label	Title	Revisions
-	Policy Statement on Nuclear Power Plant Standardization [34]	September 1987
-	Policy Statement on Regulation of Advanced Nuclear Power Plants [35]	July 1994
-	Policy Statement on the Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities [36]	August 1995
10 CFR Part 52	Licenses, Certification, and Approvals for Nuclear Power Plants. [37]	January 2022
SECY-90-016	Evolutionary Light-Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements [38]	January 1990
SECY-93-087	Certification Issues and Their Relationship to Current Regulatory Requirements [39]	April 1993
ASME/ANS RA-Sa-2009	Addenda to ASME/ANS RA-S–2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications [40]	February 2009
NUREG/CR-2300, Vol. 1, Chapter 7	PRA Procedures Guide [41]	January 1983
NUREG/CR-6595	An Approach for Estimating the Frequencies of Various Containment Failure Modes and Bypass Events [42]	Rev. 1, October 2004

15.5.2 UK and International Guidance available in Development of the Generic SMR-300

The principal UK guidance and RGP related to BDBA and SAA is specified in Table 5.

Table 5: UK Guidance and RGP

Label	Title	Revisions
SAPs	Safety Assessment Principles [19]	1
ONR-GDA-GD-006	Generic Design Assessment Guidance to Requesting Parties [43]	Rev. 0
ONR-GDA-GD-007	Generic Design Assessment Technical Guidance [18]	Rev. 0
NS-TAST-GD-005	Guidance on the Demonstration of ALARP [44]	June 2023
NS-TAST-GD-006	Design Basis Analysis [45]	December 2022
NS-TAST-GD-007	Severe Accident Analysis [20]	December 2022
IAEA SSR-2/1	Safety of Nuclear Power Plants: Design [46]	Rev. 1
IAEA SSG-2	Deterministic Safety Analysis for Nuclear Power Plants [47]	Rev. 1
IAEA SSG-4	SSG-4: Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants [48]	-
WENRA RHWG	Safety of new NPP designs [49]	March 2013

15.5.3 Codes and Standards for Accident Management and Emergency Preparedness

AM is expected to prevent the escalation of the initiating event to a SA, to mitigate the consequences of the accident, and to ultimately achieve a long-term stable condition for the plant.

There are five AM objectives:

- Preventing significant core damage.
- Terminating the progress of core damage once it has started.
- Maintaining the integrity of the containment as long as possible.
- Minimising releases to the environment.
- Achieving a long-term stable state.

SAs are initially managed by the facility operators using EOPs and then transition to SAMGs if core damage cannot be prevented. Further details can be found in Section 13.5 of the SMR-300 Preliminary Safety Report Framework [50].

Supporting the preparation of EP for the protection of people is among the principle aims of the SAA, as it considers significant but unlikely accidents and provides information on their progression and consequences within the facility, on-site and beyond the site boundaries.

In the US, the Emergency Planning requirements are specified in US NRC 10 CFR 50.47, Emergency Plans [51] and 10 CFR Part 50 Appendix E, Emergency Planning and Preparedness for Production and Utilization Facilities [52]. Further details on compliance with 10 CFR 50.47 will be provided in Part 5 of the Construction Permit Application for the Palisades SMR-300.

General regulations and guidance relating to AM and EP are defined in Table 6.

Table 6: Accident Management and Emergency Preparedness

Label	Title	Revisions
IAEA GSR Part 7	Preparedness and Response for a Nuclear or Radiological Emergency [53].	2015
IAEA GS-G-2.13	Arrangements for Preparedness for a Nuclear or Radiological Emergency [54].	2007
IAEA GSG-1	Criteria for Use in Preparedness and Response for a Nuclear or Radiological Emergency [55].	2011
IAEA SSG-54	Accident Management Programmes for Nuclear Power Plants [56].	2019
IAEA TECDOC-953	Method for the Development of Emergency Response Preparedness for Nuclear or Radiological Accidents [57].	1997
IAEA	The Fukushima Daiichi Accident, Non-serial Publications, Technical Volume 3/5 [58].	2015
IAEA	IAEA Report on Preparedness and Response for a Nuclear or Radiological Emergency in the Light of the Accident at the Fukushima Daiichi Nuclear Power Plant, Action Plan on Nuclear Safety Series [59].	2013
UK Government	How we regulate radiological and civil nuclear safety in the UK [60].	April 2021

In the UK, Section 5.7 of NS-TAST-GD-007 [20] provides guidance on how SAA should be used to inform the development of AM strategies and procedures, and also on preparation of emergency plans for protection of the public.

The output of the SAA should be used as an input to the Hazard Evaluation and Consequence Assessment (HECA) required under The Radiation (Emergency Preparedness and Public Information) Regulations 2019 (REPPIR) [61] which determines the extent of the detailed and outline emergency planning zones. The Approved Code of Practice for REPPIR 2019 is presented in [61].

15.5.4 CAE Summary

An understanding of UK and US codes and methodologies with relation to BDBA and SA has been developed. In addition, a review of the Regulatory Observations (ROs)/Regulatory Issues (RIs) relevant to BDBA, SAA and EP from previous GDAs, ONR 'Generic Design Assessment - Assessment of Reactors' [19], has been undertaken to make sure that RGP, Operating Experience (OPEX) and important lessons learnt are considered at this stage. This information is presented within this section and in SMR-300 Safety Concept for Severe Accidents Report [3] and SMR-300 Accident Management Program [4], therefore meeting the intent of Claim 2.1.3.1.

15.6 DETERMINISTIC ANALYSIS OF DEC-A EVENTS

Claim 2.1.3.2: Deterministic analysis of DEC-A events (beyond design basis accidents not resulting in core damage) confirms the absence of "cliff edge" effects and demonstrates that the facility can be brought into a long term safe, stable state with maintained containment functions.

UK Regulatory expectations on BDBA are summarised in the technical guide ONR-GDA-GD-007 [18]. In this context, the BDBA are recognised as DEC-A conditions, and the two nomenclatures can be considered synonymous as long as DEC-B correspond to Severe Accidents. ONR technical guide ONR-GDA-007 [18] reports the IAEA suggestions about which scenarios should be considered in this analysis:

- Initiating events that could lead to situations beyond the capability of safety systems that are designed for design basis accidents.
- Frequent design basis accidents combined with multiple failures that prevent the safety systems from performing their intended function to control the postulated initiating event.
- Credible postulated initiating events involving multiple failures causing the loss of a safety system while this system is used to fulfil its function as part of normal operation.

The approach to the identification and analysis of DEC-A events for the Generic SMR-300 is defined in the Safety Concept for Severe Accidents Report. Further detail on the identification and analysis is presented Part B Chapter 14 [12].

As stated in the technical guide ONR-GDA-007 [18], events classification should not be limited to Design Basis Events (DBEs). Criteria needs to be established for what scenarios should be identified for DEC-A analysis. The expectation is that events and sequences are chosen to allow the demonstration of the full extent of the defence-in-depth and capability included in the design. The criteria adopted should include a frequency cut-off but should not be limited to this. As a general approach, the PSA analysis results should be considered to investigate those sequences which were not covered by the deterministic and conservative analysis undertaken for the design basis. Engineering judgment, operating experience and international codes and standards should also be considered as identification criteria.

Once the DEC-A events have been identified, these will need to be analysed with appropriate methods and compared against the acceptance criteria. SSG-2 [47] states that the requirements on the selection, validation and use of computer codes specified for design basis accidents should apply in principle for analysis of DEC-A events. It also states that a best estimate code combined with conservative boundary conditions and assumptions (or best estimate plus uncertainty approaches) consistent with those used for DBA can be used. It concedes that best estimate analysis without quantification of uncertainties may also be used, if adequate margins to avoid cliff edge effects are demonstrated.

The analysis of BDBA should show the effectiveness of the safety measures against appropriate technical and radiological acceptance criteria. Analysis must demonstrate that the criteria defined are met. It is also part of demonstrating the challenges to barriers and implementation of the DiD approach.

15.6.1 Arguments and Evidence Construction over the Claim

It is recognised that identification of BDB faults and definition of the list of DEC-A and DEC-B events is an ongoing process throughout all stages of plant design and site-specific adaptation. Nevertheless, an indicative list of BDB events has been documented for the SMR-300 [62]. On the other hand, SA scenarios are dependent on the Faults identified, feedback from Level 2 PSA, as well as containment performance analysis [3].

At this stage of GDA, with the absence of Level 1 and Level 2 PSA, the indicative nature of the current list of BDB events and lack of subsequent analysis, Claim 2.1.3.2 cannot yet be fully demonstrated. However, arguments and evidence are set out in this section to present the approach to claim demonstration.

A representative list of initiating events is produced considering design basis characteristics and requirements. This list is then linked to the fundamental safety functions and identifying a plant response. In addition, safety demonstration must ensure there are no deviations in the design basis parameters which could threaten plant safety, and these deviations are in the range of uncertainty. This process is ensuring there are no ‘cliff-edge’ effects which are a risk to the design.

Evidence to support the claim is presented below each of the arguments in the highlighted deliverables. As outlined in the text, not all deliverables have been produced at this stage of safety case development. Where it is deemed to be outside of normal business with regards to safety case development, a GDA commitment will be raised to capture the work.

The UK Office for Nuclear Regulation (ONR) emphasises a goal-setting and non-prescriptive regulatory framework, which aligns with the argument for a holistic approach to identifying faults and challenges to Fundamental Safety Functions. This approach is rooted in the ONR's Safety Assessment Principles (SAPs) and Technical Assessment Guides (TAGs), which advocate for defence-in-depth strategies and comprehensive safety case development.

In the context of DEC-A, deterministic analysis plays a critical role. DEC-A events, which are beyond design basis accidents without core damage, are analysed to ensure that safety margins are maintained and "cliff edge" effects are avoided. This aligns with ONR's expectations for robust safety analysis, as outlined in international standards like IAEA SSR-2/1 and WENRA safety reference levels.

A holistic approach addresses the following key areas:

- Fault Identification: Systematic identification of potential faults and their impact on safety functions, ensuring that all layers of defence are considered.
- Safety Function Challenges: Evaluating how faults challenge Fundamental Safety Functions, such as reactivity control, heat removal, and containment integrity.
- Deterministic Analysis: Using conservative assumptions to confirm that the facility can transition to a long-term safe and stable state, maintaining containment functions even under DEC-A scenarios

This approach ensures that the facility's design and operational strategies are resilient, meeting ONR's regulatory expectations for nuclear safety.

2.1.3.2 - A1: A holistic approach for identification of faults and challenges to the Fundamental Safety Functions is applied.

Evidence to support the argument is given in the below deliverables:

- SMR-300 Initiating Events (IEs) list [3], [28], [62] .
    - Provides a preliminary list of IEs considered in the design, on which basis the safety demonstration is constructed. This is recognised as an initial step towards structuring a comprehensive list of IEs. This initial list will be subject to an iterative process where input from deterministic analysis will complement and justify whether a certain IE is reasonable and results in risk to the core.
- Codes and Standards Report [63].
    - Provides regulations and methodologies which give standards and regulatory expectations to the process of identification of IEs. The report outlines the codes and standards dedicated to this process, ensuring thorough and accurate assessments in line with international regulatory frameworks taking into account design specific features. The codes and standards outlined in the report provide a robust framework for the identification of IE in nuclear safety. By following these guidelines, regulatory authorities and safety assessment practitioners can conduct and review the safety approach and ultimately enhancing the safety of nuclear power plants.
- SMR-300 Safety Concept for Severe Accidents Report [3].
    - The document describes the structured safety approach aimed to ensure that the fundamental safety functions can be maintained without interruption throughout the operating life of the plant. The report addresses DEC-A events as well as providing links to Engineered Safety Features and systems in the design.
- Fault Studies Topic Area.
    - Fault Studies provide a holistic approach to the identification of faults and subsequent challenges through the production and assessment of the Consolidated Fault List (CFL). The list of faults built from OPEX and hazard identification practices are assessed via the production of a fault schedule which identifies challenges requiring assessment within DEC-A topic area. At this stage, the CFL which forms part of the Preliminary Fault Schedule (PFS) includes a preliminary list of DEC-A events for ‘in-reactor’ events [62]. Further work is required post Step 2 to validate and expand the scope. Further information on this process is captured within Chapter B14 [12].

2.1.3.2 – A2: Initiating events (IEs) identification and grouping within the scope of Level 1 PSA supports the list of IEs relevant to each operating mode for SAA.

The argument regarding the identification and grouping of IEs within the scope of Level 1 PSA focuses on ensuring that all potential initiating events are systematically identified and categorized to support severe accident analysis for each operating mode of a nuclear facility.

Below the key areas are outlined:

- Initiating Events Identification: The process begins with identifying events that could disturb the normal operation of the plant and potentially lead to core damage. These events are categorized based on their nature, such as internal events (e.g., equipment failures, human failures) and external hazards (e.g., earthquakes, floods).
- Grouping of IEs: Once identified, IEs are grouped based on their characteristics and impact on the plant's safety functions. This grouping simplifies the analysis by reducing the number of scenarios to be evaluated while ensuring that all significant events are considered.
- Relevance to Operating Modes: The analysis considers different operating modes of the plant, such as full power, low power, and shutdown states. Each mode has unique challenges and vulnerabilities, so the list of IEs is tailored to reflect these differences.
- Support for Severe Accident Analysis: The grouped IEs form the foundation for the Level 1 PSA, which evaluates the likelihood of core damage and identifies weaknesses in the plant's design or operation. This information helps in developing strategies to mitigate risks and enhance safety with regards to SAA.

This approach aligns with international standards and regulatory expectations, ensuring a comprehensive and systematic evaluation of potential risks.

Evidence to support the argument is given in the below deliverables:

- Codes & Standards Report [63].
  - Provides regulations and methodologies which give standards and regulatory expectations to the process of identification of IEs. The report outlines the codes and standards applicable to the process of identifying IEs, sequences from Level 1 PSA resulting in core damage and how these sequences should be treated further. This process is to be considered for each defined operating mode. A list of guidance is provided in the report to demonstrate the adequate approach for PSA is considered. This guidance is well established and approved through the years and industry.
- SMR-300 Safety Concept for Severe Accidents Report [3].
  - This report outlines modes of operation utilised in the safety assessment of SMR-300 with a list of IEs provided for each mode. Due to design and safety assessment maturity, the list is considered preliminary at this stage and will be finalised and ratified by SMR Level 1 PSA post PSR Rev 1. Identification of SSCs required to fulfil the fundamental safety functions is presented in the report. Once sequences leading to core damage are identified, creating bridge trees, or grouping, is the next step to the transitions from Level 1 to Level 2 PSA and grouping of sequences for further safety consideration in Level 2 PSA.
- SMR-300 Level 1 PSA [Scheduled Post PSR Rev 1].
  - It is expected a full-scope PSA will be developed based on the PSA assessment requirements and this will expand and clarify the list of IEs. Performing a comprehensive and consistent with the design Level 1 PSA will provide information about the scenarios and sequences leading to core damage.

2.1.3.2 – A3: Deterministic scenarios are defined based on the IEs and those resulting from the bridge trees at the link between Level 1 and Level 2 PSA.

The argument regarding defining deterministic scenarios based on IEs and those resulting from bridge trees at the interface between Level 1 and Level 2 PSA focuses on ensuring a seamless transition between these two levels of analysis. Key areas are outlined below.

- IEs and Bridge Trees/grouping:
  - IEs are systematically identified and grouped during the Level 1 PSA to evaluate the likelihood of core damage.
  - Bridge trees serve as a linking mechanism, translating the outcomes of Level 1 PSA (e.g., plant damage states) into inputs for Level 2 PSA, which focuses on containment performance and radiological release.
- Defining Deterministic Scenarios:
  - Deterministic scenarios are developed by postulating specific sequences of events based on the identified IEs and their progression through the bridge trees.
  - These scenarios consider the status of safety systems, operator actions, and environmental conditions to evaluate the plant's response to potential accidents.
- Integration with Level 2 PSA:
  - The deterministic scenarios provide a structured framework for analysing severe accident progression, including containment challenges and potential radiological releases.

This integration ensures that the insights from Level 1 PSA are effectively utilised in Level 2 PSA, enhancing the overall safety assessment.

Evidence to support the argument is given in the below deliverables:

- SMR-300 Safety Concept for Severe Accidents Report [3].
  - Provides main plant characteristics related to plant safety and DEC-A events, modes of operation and available Engineered Safety Features (ESFs), operating parameters and limitations. The report presents the main aspects of identification of deterministic scenarios for calculating accident progression into a severe stage together with phenomena, which are relevant per accident phases and possible mitigation measures. As a result of this, deterministic analyses are properly defined. Additional deterministic analysis is also considered as part of demonstration of the ultimate containment capacity.
- SMR-300 Accident Management Program [4].
  - The report outlines the key elements that the team responsible for preparing, developing, and implementing a plant-specific Accident Management Program (AMP) at a nuclear power plant should address in response to DEC-A events. It provides guidance and a basis for further AM development, up to the stage where site-specific conditions have impact on the matter. The AM report focuses on the fourth level of defence-in-depth, including transitions from the third level and into the fifth level. It follows best practices in accident

management and emergency preparedness development and represents an integral part of the ALARP demonstration.

- SMR-300 Level 1/2 PSA [Scheduled Post PSR Rev 1].
    - Level 1 and Level 2 PSA which include all modes of operation, all system configurations and potential sources of radioactive releases outside the site for all corresponding IEs will be developed post PSR Rev 1. It is expected that Level 2 PSA will be developed, and the respective bridge trees accounted for. This will give additional scenarios which are considered possible from PSA point of view giving completeness to the safety assessment and will set the sequence of phenomena during SA progression.

15.6.2 CAE Summary

As outlined within this section, not all supporting evidence has been produced to fully meet this claim. An understanding of relevant faults and initiating events has been developed utilising extant evidence as well as the development of procedures that will be applied. The approach to the identification and assessment of DEC-A events at PSR Revision 1 is based upon the available Preliminary Fault Schedule (PFS) [62] from Fault Studies Topic Area and is referenced in the deliverable Safety Concept for Severe Accidents Report [3] and SMR-300 Accident Management Program [4].

The main sources of information about the range of initiators, plant systems operation and safety challenges are provided from the Fault Studies area and Level 1/2 PSA leading to 'cliff-edge' effect accounting.

At present, without a SMR-300 Level 1/2 PSA or SA deterministic analysis, it is challenging to demonstrate with confidence that the facility can be brought into a long term safe, stable state during a DEC-A scenario. As the safety assessment to support the SMR-300 matures, it is expected that this demonstration will be possible in the future.

The key areas of work to be conducted to support each argument are outlined above. Once all source documentation has been developed, the above processes will be conducted, and the claim met.

The below GDA commitment has been raised to capture outstanding work in support of this claim.

C_SAA_084: Further UK-based safety analysis is required to support the development of a comprehensive deterministic analysis of DEC A events and confirm the absence of 'cliff edge' effects. A Commitment is raised to conduct deterministic analysis of DEC A events.

Target for Resolution - Issue of UK Pre-Construction SSEC.

15.7 DEMONSTRATION OF PLANT RESPONSE TO DEC-B EVENTS

Claim 2.1.3.3: Severe accident analysis of DEC-B events (beyond design basis accidents with core damage) demonstrates that the facility can be brought into a long term safe, stable state with maintained containment functions.

As stated in the SAPs [19], when an initiating event is highly unlikely and difficult to predict, it is not always reasonably practicable to incorporate conservatively designed safety measures expected for such events. It is reasonably practicable though to plan for how events with more severe consequences than allowed for in the design basis would be managed and to provide equipment and procedures that would be needed to control or mitigate their consequences.

The SAPs [19] base the definition of SA on numerical targets of radiological consequences. As stated in TAG NS-TAST-GD-007 [18], any event that could reasonably exceed any of these numerical targets is a potential SA and should be considered in the safety case. In the UK GDA context, the SAA can be considered consistent with the IAEA [64] and WENRA [49] expectations for deterministic analysis of DEC associated with core damage (DEC-B analysis). The objectives for the Generic SMR-300 as defined in TAG NS-TAST-GD-007 [18] are:

- Demonstrate an understanding of phenomena and risks.
- Demonstrate DiD.
- Identify performance and environmental qualification requirements for equipment.
- Identify mission times and stocks of inventories.
- Demonstrate learning from Fukushima and other accidents.
- Support PSA modelling.
- Inform emergency procedures.
- Demonstrate that risks are ALARP.
- Demonstrate that large or early releases have been practically eliminated.

The approach to the identification and assessment of DEC-B events for the Generic SMR-300 is defined in the Safety Concept for Severe Accidents Report [3]. The aim of the report is to provide information about SMR-300 design specifics related to SAs, phenomena applicable during SA progression as well as the technical means in place (SSCs and ESFs) to cope with them. This is part of the overall demonstration of DiD and analysis for safety substantiation claims.

At the PSR stage, the information is based on identification of sequences and events leading to DEC for SMR-300. This is based on available PSA data and stand-alone safety justifications of ESFs and SSCs. Requirements for analysis and phenomena assessment will be part of the report and will be supporting further PSR chapter development, including items such as In-vessel retention (IVR), High Pressure Melt Ejection (HPME), etc including phenomena assessment. It should be recognised that, while this report is delivered within SAA, substantial input from other topics such as Fault Studies, PSA, HF and others will be used.

15.7.1 Arguments and Evidence Construction over the Claim

As we consider DEC-B events, that possess requirements on the last barrier against radioactivity release to the environment, the focus is on the containment and its performance and response to the phenomena which challenge its integrity and functions.

At this stage of GDA, with the absence of Level 2 PSA and the indicative nature of the current IEs, Claim 2.1.3.3 cannot yet be fully demonstrated. However, arguments and evidence are set out in this section to present the approach to claim demonstration.

Evidence to support the claim is presented below each of the arguments in the highlighted deliverables. As outlined in the text, not all deliverables have been produced at this stage of safety case development. Where it is deemed to be outside of normal business with regards to safety case development, a GDA commitment will be raised to capture the work.

2.1.3.3 – A1: IEs identification and grouping in the scope of Level 2 PSA supports the list of deterministic analysis relevant to assessment of ultimate capacity of the containment.

The identification and grouping of IEs within the scope of Level 2 PSA play a crucial role in supporting deterministic analyses, especially when evaluating the ultimate capacity of the containment. By systematically identifying and categorising IEs, the analysis ensures that all potential challenges to containment integrity during severe accidents are accounted for.

This process begins with a comprehensive review of both internal events, such as reactor system failures, and external hazards, like natural disasters. The IEs are then grouped based on their characteristics and potential to escalate into scenarios that may impact containment performance. These groups form the basis for defining deterministic scenarios.

Deterministic scenarios, grounded in the grouped IEs, allow for a detailed assessment of containment performance under severe accident conditions. Key factors such as pressure, temperature, and structural stresses are analysed to determine whether the containment structure can withstand such extreme scenarios. The evaluation also considers phenomena like hydrogen generation, core-concrete interaction, and heat loads, ensure that safety margins are maintained.

Ultimately, this approach not only confirms the containment's ability to resist failure but also aligns with international regulatory standards. The insights gained are invaluable for informing design enhancements, operational strategies, and emergency preparedness, thus bolstering the facility's overall safety.

Evidence to support the argument is given in the below deliverables:

- SMR-300 Safety Concept for Severe Accidents Report [3].
    - The safety concept report has been developed to summarise and describe the main objectives and strategy for severe accident (DEC-B) management including DiD principles in order to achieve a safe state for the plant. This includes an indicative list and grouping of DEC-B IEs and the subsequent approach to conducting SA deterministic analysis post PSR Rev 1. The report explains how the SAs are accounted in the overall safety strategy and

mitigation of potential consequences. It contains also the SSCs which play role in the process, their capacity and principle of operation.

- SMR-300 Accident Management Program [4].
    - Provides a structured approach to assess scenarios for deterministic calculations which involve phenomena challenging the last barrier. Once scenarios are identified, a general AM program is to be structured in order to identify potential dedicated and non-dedicated systems which could be used in AM strategies.
- Level 2 PSA (scheduled post PSR Rev 1).
    - Level 2 PSA will be developed, and respective bridge trees accounted. This will give additional scenarios which are considered possible from PSA point of view and will set the sequence of phenomena during SA progression. Level 2 PSA is one of the main sources of scenarios leading to core damage and challenging containment.

2.1.3.3 – A2: Deterministic analysis demonstrating efficiency and ESF implementation though the Severe Accident Management (SAM) strategies (analysis with and without accident management measures).

A major role in safety demonstration and accident management, is the operability and efficiency of the ESFs. These analyses are the basis of accident management strategies development and achieving safety goals related to the SAM objectives. Due to the design stage and the GDA approach, such analysis is planned for later stage but the expectations and objectives as summarised in the supporting deliverables. Evidence to support the argument are given in the below deliverables:

- SMR-300 Safety Concept for Severe Accidents Report [3].
    - Provides main plant characteristics related to plant safety and DEC-A and DEC-B events, modes of operation and available ESFs, operating parameters and limitations. Deterministic analysis plays a key role in demonstrating the effectiveness of ESF and SAM strategies in mitigating severe accident scenarios. A Safety Concept Document built around this argument highlights the facility's ability to handle extreme events and maintain safety.
    - The deterministic analysis will evaluate accident scenarios both with and without the implementation of SAM strategies. By doing so, it clearly illustrates the impact of these strategies in preventing or mitigating severe accident consequences. This includes demonstrating the timely and reliable activation of ESF, which preserves critical safety functions such as heat removal and containment integrity.
    - Through this comparison, the analysis shows how SAM strategies reduce risks and enhance the overall safety of the facility. It also provides evidence that the containment structure and safety systems can effectively manage the conditions of severe accidents, maintaining integrity and avoiding catastrophic outcomes.
    - The Safety Concept Document aligns with international regulatory standards, such as those set by the IAEA and WENRA, reinforcing stakeholder confidence in the facility’s commitment to safety. It serves as a comprehensive safety case,

showcasing the robustness of accident management measures and their critical role in ensuring the facility's resilience and the protection of public and environmental health.

- SMR-300 Accident Management Program [4].
  - Provides a structured approach to assess scenarios for deterministic calculations which involve phenomena challenging the last barrier. The AMP plays a critical role in ensuring the safety and resilience of nuclear facilities by mitigating the consequences of severe accidents. It is designed to prevent the escalation of such events through specific measures, such as reactor depressurisation, core cooling, and containment venting.
  - AMP enhances preparedness by equipping operators with well-defined strategies and procedures to respond effectively during severe accidents. This includes comprehensive training, regular drills, and clear decision-making guidelines to manage high-stress situations.
  - The program also ensures the efficient use of ESF to preserve vital safety functions like heat removal and containment integrity. By managing severe accident conditions effectively, AMP minimises the likelihood of radiological releases and ensures the facility transitions into a safe, stable state.
  - Additionally, AMP fosters public and stakeholder confidence by demonstrating the facility's commitment to safety and compliance with international standards. It incorporates continuous improvement by learning from industry experiences and research, ensuring it remains adaptive to emerging challenges.
  - In essence, the AMP is a cornerstone of nuclear safety, providing robust tools and strategies to protect both people and the environment.
- SA Deterministic Analysis (Scheduled Post PSR Rev 1).
  - SA Deterministic analysis plays a fundamental role in ensuring the safety and reliability of nuclear facilities by providing a structured and comprehensive evaluation of potential accident scenarios. Its primary contribution lies in systematically assessing how a facility's systems and structures perform under various predefined conditions.
  - This type of analysis enables the identification of vulnerabilities in the plant's design or operation during SA, ensuring that safety margins are maintained across all scenarios. It evaluates the effectiveness of safety systems, such as ESF, in protecting critical safety functions like reactivity control, heat removal, and containment integrity in the prevention and mitigation of SA. By simulating different accident conditions, deterministic analysis confirms that these systems perform as intended and that no "cliff edge" effects jeopardize the plant's safety.
  - Additionally, deterministic analysis provides the foundation for developing and validating SAM strategies. It helps to define and analyse accident scenarios both with and without the implementation of these strategies, demonstrating their effectiveness in mitigating potential consequences and ensuring a stable, safe state for the facility.
  - SA deterministic analysis supports compliance with international safety standards, providing clear evidence of the plant's capability to handle challenging scenarios. This not only reinforces public and stakeholder

confidence but also informs continuous improvement in plant design and emergency preparedness.
- Ultimately, SA deterministic analysis ensures a robust safety framework, helping to safeguard both the environment and human health while enhancing the resilience of nuclear safety.
- These are deterministic analyses which are part of the SAMG and procedures preparation. Their aim is to target the containment structure and included containment performance analysis and decision making in accident management that support anaylsis.

15.7.2 CAE Summary

As outlined, not all supporting evidence has been produced to fully meet this claim. An understanding of relevant faults and initiating events has been developed utilising extant evidence as well as the development of procedures that will be applied.

Severe Accidents classified as DEC-B events, require a specific approach to assess and interpret the results. Scenarios themselves are dependent on outcomes of the Level 1/2 PSA and additional containment deterministic analysis, as well as any additional analysis demonstrating challenges to the containment by different accident phenomena and accident conditions. At present, without a SMR-300 Level 1/2 PSA or SA deterministic analysis, it is challenging to demonstrate with confidence that the that the facility can be brought into a long term safe, stable state with maintained containment functions during a DEC-B scenario. As the safety assessment to support the SMR-300 matures, it is expected that this demonstration will be possible in the future.

The objectives and approach to assessing DEC-B events and accident phenomena is understood and presented in both this section, SMR-300 Safety Concept for Severe Accidents Report [3] and SMR-300 Accident Management Program [4]. Once supporting source documentation is available, the above processes will be conducted, and the claim met.

The below GDA commitment has been raised to capture outstanding work in support of this claim.

C_SAA_085: Further UK-based safety analysis is required to support the development of a comprehensive deterministic analysis of DEC B events and confirm the absence of ‘cliff edge’ effects. A Commitment is raised to conduct deterministic analysis of DEC B events.

Target for Resolution - Issue of UK Pre-Construction SSEC.

15.8 SAFETY MEANS FOR ACCIDENT CONTROL AND MITIGATION

Claim 2.1.3.4: Additional reasonably practicable safety functions and safety measures are identified, categorised and classified based on their importance to nuclear safety for the purpose of Severe Accident management.

The SMR-300 Generic Design has a variety of ESFs, strategies and procedures for responding to DBA, BDBA, and SAs. This sub-chapter deals with the ESFs dedicated to prevention and mitigation of SAs.

Description of the systems and their modes of operation is provided in PSR Part B Chapter 1 [9].

Based on the current design, a preliminary list of SSCs that are available for Severe Accident Mitigation is presented in Table 7, from [28].

Table 7: SSCs for Severe Accident Mitigation

Function	Systems/Components	Actions
Reactor Vessel Overpressure Protection	Pressuriser (PZR) Safety Valves	Depressurise the reactor vessel
	Safety valve in the Residual Heat Removal System (RHR)	
	Automatic Depressurization System (ADS)	
	Reactor Pressure Vessel (RPV) head vent valves	
External Reactor Vessel Cooling	Passive Core Make Up Water System Water Tanks (PCMWTs) of the Passive Core Make Up Water System (PCM)	Remove heat from the reactor vessel exterior surface.
Containment Structure (CS) Integrity	Containment Structure (CS)	Provide containment pressure capacity, large volume to power ratio.
Containment Isolation	Containment Isolation System (CIS)	Close normally open pathways through the containment envelope.
Hydrogen Control	Containment Ventilation System (CBV)	Mix the containment atmosphere.
	Combustible Gas Control System (CGC)	Remove hydrogen.
Containment Cooling	Containment Building Ventilation (CBV)	Remove heat from the containment to the Chilled Water System.
	Passive Containment Heat Removal System (PCH)	Remove heat from the containment to the ultimate heat sink.
Ex-Vessel Core Debris Cooling	Containment floor flooding by water from PCMWTs of the PCM	Flooding the debris.
Containment Overpressure Protection	CBV purge function	Controlled depressurisation of the containment.
Flexible Coping Strategies (FLEX) Containment Water Addition from external source	Residual Heat Removal System (RHR) FLEX connections	Make-up volume for containment, such as may be needed for substantial containment bypass.
FLEX Auxiliary Power from external source	Standby Diesel Generator System (outside scope of SSEC)	Installed reliable non-safety related AC power supply to restore plant ac power for extended loss of off-site power.
	Plant Non-1E Power Distribution System FLEX connections	Restore plant AC power for extended loss of off-site power.

Further information on the modelling of SAs and the plant response to core damage events is presented in [3] and [4].

15.8.1 Arguments and Evidence Construction over the Claim

The Holtec SMR-300 plant safety functions are arranged hierarchically to manage the complexity and facilitate the analysis, implementation, maintenance, and communication. Further discussion on US and UK safety functions, category and system classification is in Chapter B14 [12].

The mitigation of severe accidents is typically classified as Safety Function Category (SFC) B or SFC-C, recognising that the frequency of demand on such safety function demands should be very low, and hence their importance to safety is reduced compared to prevention and mitigation of faults and hazards. Furthermore, in the absence of full unmitigated consequence assessment, it has been assumed that core damage accidents associated with DEC events lead to an off-site dose in excess of 100 mSv, as justified in Section 7.0 of Safety Assessment Handbook [65]. This will be reviewed at the Pre-Construction Safety Report (PCSR) stage when more detailed consequence assessment will be available.

Where passive safety systems and engineered safety measures cannot deliver a safety function, reliance may be placed on humans to perform safety related actions. Where it is necessary to place reliance on humans, a proportionate level of HF assessment analysis is to be performed in accordance with ONR SAPs [19] and detailed sources of guidance on UK HF good practice, such as published regulatory guidance (e.g. ONR TAGs, Health and Safety Executive guidance).

2.1.3.4 – A1: Performed deterministic analysis as well as those dedicated to assessment of the containment ultimate capacity allow the identification of positive and negative effects in implementation of SAM (Optioneering and containment performance demonstration).

This argument relates to identifying, categorising, and classifying additional reasonably practicable safety functions and measures for nuclear safety highlighting a proactive approach to enhancing accident mitigation, especially in the context of Severe Accident Management (SAM). This approach aligns with international safety standards and demonstrates a commitment to continuously strengthening nuclear safety. By integrating AMP, Level 2 PSA, and deterministic analysis, the facility ensures a comprehensive and resilient safety framework capable of effectively managing severe accidents. Below is outlined how this applies to the implementation of the AMP, Level 2 PSA, and deterministic analysis:

- SMR-300 Accident Management Program [56].
    - Provides a structured approach to assess scenarios for deterministic calculations which involve phenomena challenging the last barrier. The AMP benefits directly from identifying and implementing additional safety functions and measures. These include improvements such as enhanced containment cooling, hydrogen management systems, or robust containment venting mechanisms.

    - By outlining the process for categorising and classifying these measures based on their importance to nuclear safety, the AMP ensures prioritisation and effective deployment during severe accidents.
    - The integration of these measures into SAM strategies strengthens the facility's ability to mitigate accident consequences and transition to a safe, stable state.
- Level 2 PSA [Scheduled Post PSR Rev 1].
    - It is expected that Level 2 PSA will be developed, and respective bridge trees accounted. This will give additional scenarios which are considered possible from PSA point of view and will set the sequence of phenomena during SA progression. Level 2 PSA evaluates the effectiveness of safety systems in preventing radiological releases during severe accidents. By incorporating additional safety measures, the PSA becomes more comprehensive and reflective of the facility's capabilities.
    - The classification of measures allows Level 2 PSA to assess their probabilistic contribution to reducing the likelihood and impact of containment failures, enhancing the overall safety profile of the plant.
- SA Deterministic Analysis [Scheduled Post PSR Rev 1].
    - These are deterministic analysis which target the containment structure mainly. These are containment performance analysis. This includes assessment of resulting pressure in case of 100% Zr oxidation and theoretical combustion.
    - Deterministic analysis ensures that the additional safety functions and measures can withstand extreme accident scenarios, confirming their ability to maintain critical safety functions like containment integrity.
    - It provides a structured approach to validate the performance of these measures, even under highly challenging conditions, thereby reinforcing the robustness of the safety case.
    - The analysis also identifies potential vulnerabilities and informs necessary design or procedural improvements.

15.8.2 CAE Summary

As outlined above, not all supporting evidence is available to fully meet this claim. The SMR-160 Design Standard for Severe Accident Requirements [28] has been utilised to produce an indicative list of SSC/ESFs that support the prevention and mitigation of severe accidents. However, as SMR-300 documentation and safety analysis is produced and assessed an SMR-300 list of SSCs/ESFs relevant to the prevention and mitigation of severe accidents will be produced. From this list, additional safety functions and safety measures will be identified, categorised and classified based on their importance to nuclear safety for the purpose of Severe Accident management.

As detailed above and in the AMP, the process and approach to meet this claim is understood. However, without a Level 2 PSA and subsequent deterministic analysis, this claim cannot be met. Once supporting documentation is developed, the analysis will be conducted to fully meet this claim.

The below GDA commitment has been raised to capture outstanding work in support of this claim.

C_SAA_086: Further UK-based safety analysis is required to support the categorisation and classification of any additional safety functions and safety measures required for accident management. A Commitment is raised to conduct further analysis with regards to any additional safety functions or safety measures required for accident management.

Target for Resolution: Issue of UK Pre-Construction SSEC.

15.9 ACCIDENT MANAGEMENT AND EMERGENCY PREPAREDNESS

Claim 2.1.3.5: Accident management and emergency preparedness take all reasonably practicable measures to prepare for possible accidents, and to mitigate their consequences should they occur.

Holtec SMR-300 design should demonstrate adequate and efficient response to BDBA, their progression, and mitigation of consequences. Therefore, an Accident Management Program Report [4] was submitted, as a supporting deliverable to Chapter 15. The report outlines the key elements that are needed for preparing, developing, and implementing a plant-specific Accident Management Program .

Although they are unlikely to be needed, SAM programmes are a critical part of the DiD concept, which is a hierarchical deployment of different levels of equipment and procedures in a graded approach to protect against a wide variety of incidents, accidents, equipment failures, human failures and events initiated outside the plant. In general, severe accident management programs are designed to:

- Evaluate generically the capability of existing plants to tolerate a SA.
- Identify events that can lead to SAs and formulate preventive and mitigation strategies.
- Identify short-term and long-term measures for handling SAs.

Paragraph 776 of the SAPs [19] states that accident management should be based on the facility's SAA. The SAA (supported by aspects of the PSA) should provide the following:

- A representative selection of initial accident states.
- Appropriate points for the transition into AM guidelines (criteria for entry into the SA domain).
- The symptoms that will allow the operators to identify the true state of the plant.
- Alternative scenarios for how accident sequences might progress and an analysis of the likely effectiveness of different strategies for these.
- The plant monitoring functions that are required to support the delivery of the SA measures.
- The required plant and equipment (including mobile equipment) and the associated human actions to deploy these.
- The timescales for key operator actions.
- Environmental conditions within and around the plant and the effect on deployment of equipment and people.
- Defining mission times for plant and equipment and the supply of consumables.
- Expected releases.

To protect the public and the environment from the consequences of a NPP accident, each plant operator establishes a SA management programme, which is kept under constant review and development. The main objective of the guidelines used to design such programmes is to utilise any available equipment and ESFs at the NPP to terminate core damage, maintain containment integrity and minimise the release of radioactivity.

In general, NPPs are equipped with multiple safety systems able to deal with a wide range of abnormal operating conditions. They also have well-proven EOPs that help operators achieve a stable and safe end state. However, the most severe circumstances can result in damage to the nuclear fuel and the containment structures, with a potential to release radioactivity to the environment. Even in these events, the consequences can still be mitigated using available and, in some cases, dedicated plant equipment.

The AMP report is focused on providing guidance and basis for next step up to the stage where site dependent conditions are considered. The report contains SSCs which are credited with accident prevention, control and management with a link to their detailed description and safety effectiveness demonstration. ESF operation and effectiveness are highlighted and linked to strategies and procedures which will be needed for a complete AM plan. In addition, some requirements and features of the emergency centre, control room, Post Accident Monitoring System, etc are presented.

15.9.1 Arguments and Evidence Construction over the Claim

It is recognised plant procedures and guidelines would be developed completely at the site stage and continue development after commissioning of NPP. Nevertheless, in order to enter commercial operation, a NPP should be prepared at all levels of DiD.

At the current stage of the GDA, a single argument is defined which reflects safety expectations with respect to accident management. This argument refers to the main data and source of information needed for further AM development and is subject to further development when site Hazards are defined as well.

Evidence to support the claim is presented below each of the arguments in the highlighted deliverables. As outlined in the text, not all deliverables have been produced at this stage of safety case development. Where it is deemed to be outside of normal business with regards to safety case development, a GDA commitment will be raised to capture the work.

2.1.3.5 – A1: Deterministic analysis verifies that emergency preparedness actions result in minimum releases to the environment (use of source term).

The claim that "Accident management and emergency preparedness take all reasonably practicable measures to prepare for possible accidents and to mitigate their consequences should they occur" emphasises the importance of a proactive, well-structured approach to ensuring nuclear safety. When viewed through the lens of deterministic analysis combined with emergency preparedness actions, this claim further reinforces the facility's ability to minimise releases to the environment effectively. The AMP, Level 2 PSA, and Level 3 PSA each play distinct yet interconnected roles in ensuring nuclear safety by addressing accident prevention, mitigation, and consequence management.

- SMR-300 Accident Management Program [4].
  - The report outlines accident management legal framework, objectives, principles and the key elements that are needed for preparing, developing, and implementing a plant-specific AMP Program at a Nuclear Power Plant. It provides a structured approach to assess scenarios for deterministic

calculations which involve phenomena challenging the last barrier. The AMP focuses on implementing measures and strategies to mitigate the consequences of severe accidents within the plant. It includes actions such as core cooling, containment venting, and hydrogen management. AMP equips operators with the necessary tools, training, and clear procedural guidelines to stabilize the facility during emergencies. Its primary goal is to prevent accidents from escalating and to maintain containment integrity, ensuring the plant transitions to a safe and stable condition.

- Level 2 PSA [Scheduled Post PSR Rev 1].
  - It is expected that Level 2 PSA will be developed, and respective bridge trees accounted. This will give additional scenarios which are considered possible from PSA point of view and will set the sequence of phenomena during SA progression. The Level 2 PSA extends the analysis by evaluating severe accident progression after core damage. It assesses the likelihood and impact of containment failure modes by considering plant damage states from Level 1 PSA. Level 2 PSA helps identify vulnerabilities in the containment system, evaluates the effectiveness of additional safety measures, and ensures that the containment's performance meets safety expectations. Its main focus is to reduce the probability of significant radiological releases.
- Level 3 PSA [Scheduled Post PSR Rev 1].
  - It is expected a comprehensive analysis covering the source term and releases assessment with potential dose load will be developed. The Level 3 PSA examines the broader off-site consequences of radioactive releases. It analyses the transport and dispersion of radionuclides into the environment, evaluating their impact on public health and safety. Level 3 PSA also assesses the effectiveness of emergency preparedness actions, such as evacuation plans and protective measures, to minimise radiological exposure. It further quantifies potential societal and economic consequences, providing valuable insights for optimising emergency response and recovery strategies.

15.9.2 CAE Summary

As outlined above, not all supporting evidence has been produced to meet this claim. Accident management and emergency preparedness plans will be developed for site specific scenarios, therefore, cannot be produced at this stage. The theory and requirements of the AMP are understood and will be developed alongside safety analysis (Level 2 and 3 PSA) as the design and location of the facility are developed and understood. Utilising the above evidence, when available, under the guidance of the AMP will ensure this claim is met.

The below GDA commitment has been raised to capture outstanding work in support of this claim.

C_SAA_086: Further UK-based safety analysis is required to support the categorisation and classification of any additional safety functions and safety measures required for accident management. A Commitment is raised to conduct further analysis with regards to any additional safety functions or safety measures required for accident management.

Target for Resolution: Issue of UK Pre-Construction SSEC.

15.10 CHAPTER SUMMARY AND CONTRIBUTION TO ALARP

This sub-chapter provides an overall summary and conclusion of the BDBA, SA and EP and how this Chapter contributes to the overall demonstration of ALARP for the generic SMR-300. Part A Chapter 5 [8] sets out the overall approach for demonstration of ALARP and how contributions from individual Chapters are consolidated.

This sub-chapter therefore consists of the following elements:

- Technical Summary.
- ALARP Summary
    - Demonstration of Relevant RGP.
    - Evaluation of Risk and Demonstration Against Risk Targets (where applicable).
    - Options Considered to Reduce Risk.
- GDA Commitments.
- Conclusion.

A review against these elements is presented below under the corresponding headings.

15.10.1 Technical Summary

PSR Chapter B Part 15 demonstrates that the BDBA, Severe Accident Analysis, and Emergency Preparedness topics will meet the high-level Claims of the SSEC and that the ESFs can be substantiated at PCSR stage. This is demonstrated through the following sub-claim:

Claim 2.1.3: Beyond design basis faults and severe accidents are appropriately identified and risk assessed to be tolerable and As Low As Reasonably Practicable (ALARP).

The key requirement of the SMR-300 is to demonstrate for DEC events and accidents that have the potential to lead to severe consequences have been systematically analysed, and the analysis is used to identify appropriate preventative and mitigating measures beyond those derived from the DBA. This will be undertaken by considering:

- BDBA (DEC-A) events and demonstrating sufficient margins exist in the design of ESFs such that core damage does not occur.
- SA (DEC-B) events and demonstrating that the ESFs provided in the design mitigate the consequences of core damage.
- An Accident Management Programme that shows the combination of ESFs and EP plans can demonstrate that the consequences of SAs are minimised and shown ALARP.

15.10.2 ALARP Summary

15.10.2.1 Demonstration of RGP

Due to the differences in regulatory approaches between the UK and US, the DEC topic is assessed in differing ways. Within the UK, a DEC systematic analysis is carried out in a way complementary to DBAA and PSA as outlined in sub-chapters 15.6 and 15.7. It is expected that deterministic analysis of design extension conditions without significant fuel damage (DEC-A) will be considered as part of the fault studies topic area due to the similarity of the codes and methods used, while DEC-B type scenarios will be assessed in the SA topic area as supporting analysis is developed post PSR Rev 1.

In the US a more prescriptive approach is taken, and DEC are analysed in the PSA domain. As the safety case matures, the SAA chapter will include more data about the design progress, safety substantiation and consideration of RGPs which will lead to any options to reduce risk.

15.10.2.2 Options Considered to Reduce Risk

No specific risk reduction options identified to date and discussed within this chapter. However, it is noted that there is a strong interface with Design Challenge Paper (DCP) 03 [66] which has recognised differences in the approach to safety assessment between UK best practice and the requirements of the NRC. The outcome of the resolution in DCP 03 will be applicable to BDBA and SAA.

15.10.3 GDA Commitments

GDA Commitments which relate to this Chapter have been formally captured in the Commitments, Assumptions and Requirements process [6] Further details of this process is provided in Part A Chapter 4.

A summary of the commitments raised from this chapter is presented below:

C_SAA_084: Further UK-based safety analysis is required to support the development of a comprehensive deterministic analysis of DEC A events and confirm the absence of ‘cliff edge’ effects. A Commitment is raised to conduct deterministic analysis of DEC A events.

Target for Resolution - Issue of UK Pre-Construction SSEC.

C_SAA_085: Further UK-based safety analysis is required to support the development of a comprehensive deterministic analysis of DEC B events and confirm the absence of ‘cliff edge’ effects. A Commitment is raised to conduct deterministic analysis of DEC B events.

Target for Resolution - Issue of UK Pre-Construction SSEC.

C_SAA_086: Further UK-based safety analysis is required to support the categorisation and classification of any additional safety functions and safety measures required for accident management. A Commitment is raised to conduct further analysis with regards to any additional safety functions or safety measures required for accident management.

Target for Resolution: Issue of UK Pre-Construction SSEC.

It is noted that these commitments are specific to this chapter and are in addition to the overarching fault studies commitment C_Faul_103 [12]. It is expected these commitments will form part of the overall resolution plan for C_Faul_103.

15.10.4 Conclusion

The conclusion of Part B Chapter 15 of the PSR is that:

- The Claims identified have been either met to a maturity aligned with the current vision of the PSR or a clear view and approach is defined to achieve and successfully demonstrate plant safety which is outlined in each CAE Summary.
- The Severe Accidents Gap Analysis Report, produced for PSR Revision 1, defines the level of potential additional work which is needed to support safety claims and meet UK regulations.
- The Safety Concept for Severe Accidents Report [3], produced for PSR Revision 1, and defines the approach to the identification and assessment of DEC events.
- The AMP defines the approach to AM and EP.
- When available, safety analysis from the Fault Studies Topic Area and PSA will be utilised to conduct safety assessments in support of BDBAs and SAs.

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15.12 LIST OF APPENDICES

Appendix A PSR Part B Chapter 15 CAE Route Map

Table 8: PSR Part B Chapter 15 CAE Route Map

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