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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, 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 Preliminary Safety Report (PSR) Part A Chapter 1 'Introduction' [1].

The Fundamental Purpose is achieved through the Fundamental Objective of the 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 Analysis (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 PSR 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 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 Generic 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 Subchapter 15.3.

Together with PSR Part B Chapter 14 'Safety and Design Basis Accident Analysis' [3], these two chapters provide 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 (SAs).

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	N/A

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.

Part B Chapter 15 covers the Codes and Standards (subchapter 15.4) and Methodologies for Beyond Design Basis Accidents (subchapter 15.5), SAA (subchapter 15.6) and AM and EP (subchapter 15.8). The identification of Engineered Safety Features (ESFs) is discussed in subchapter 15.7. At the end, a summary is provided including a Technical and ALARP summary (subchapter 15.9).

The Structures, Systems, and Components (SSCs) included within the scope of this Chapter are outlined in subchapter 15.7.

15.1.2 Assumptions

It is assumed that no specific SMR-300 assessment results will be available during the GDA.

15.1.3 Interfaces with Other PSR Chapters

The other PSR Chapters that Chapter B15 interfaces with are described below.

Part A Chapter 2 ‘General Design Aspects and Site Characteristics’ [4]: 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 DECs on Chapter B15.

Part A Chapter 5 ‘Summary of ALARP’ [5]: The chapter presents the ALARP methodology and ALARP justifications for the SMR-300, this provides the basis for subchapter 15.9 which outlines the ALARP assessment of BDBA and SA, including a description of Forward Action Plans (FAPs).

Part B Chapter 1 ’Reactor Design and Engineering Aspects’ [6] and Part B Chapter 2 ‘Reactor Fuel and Core’ [7] 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 DEC-A and SAA.

Part B Chapter 10 ‘Radiological Protection’ [8] for derivation of the operational source term noting that Chapter B15 should provide the Beyond Design Basis (BDB) and SA source terms.

Part B Chapter 14 ‘Design Basis Accident Analysis (DBAA)’ [9]: Chapter B15 extends the Design Basis Analysis (DBA) of Chapter B14 [3], which provides initiating events for sequences identification and methodology for hazard identification.

Part B Chapter 16 ‘Probabilistic Safety Analysis (PSA)’ [10]: 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)’ [11]: Chapter B17 substantiates the claims on operator actions under BDB and SA conditions.

Part B Chapter 18 ‘Structural Integrity (SI)’ [12]: 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’ [13]: Chapter B20 presents the design substantiation of civil structures. Chapter B15 may provide thermo-hydraulic conditions in internal containment under SA for design of civil structures.

Part B Chapter 23 ‘Reactor Chemistry’ [14]: Chapter B23 provides accident process 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 BDBA, SAA AND EP

The aim of PSR Part B Chapter 15 is to address the evaluation of the 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 [3].

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 [15]. In this context, the BDBA are recognised as DEC-A conditions, and the two nomenclatures can be considered synonymous. SAs (DEC-B) are discussed in subchapter 15.6. 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.

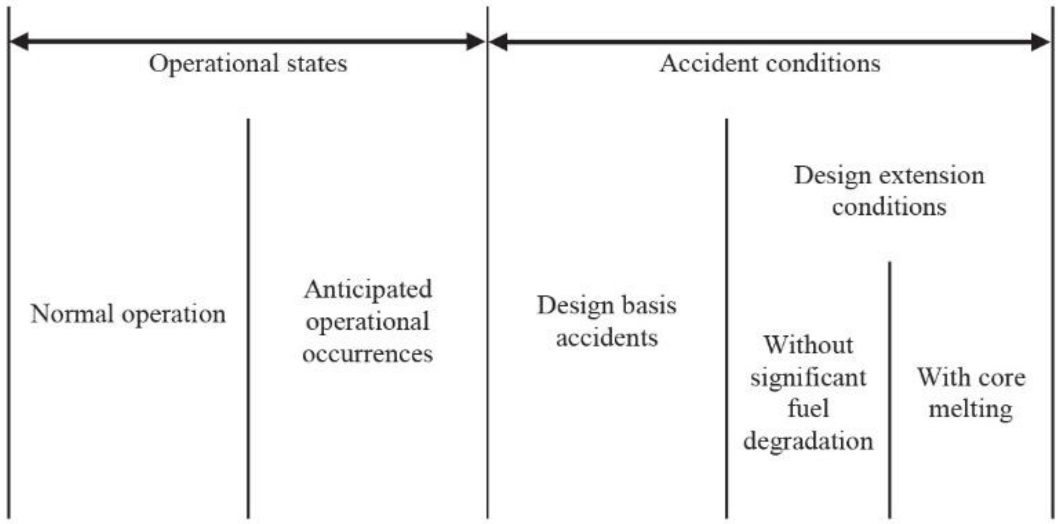

Figure 1: Plant States Considered in Design for a Nuclear Power Plant

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 align with the UK Relevant Good Practice (RGP) and the relevant Office For Nuclear Regulation (ONR) Safety Assessment

Principles (SAPs) [16] and Technical Assessment Guide (TAGs) as defined in subchapter 15.4, 'Codes and Standards'.

The DBAA of the Generic SMR-300 is addressed in Part B Chapter 14 'Safety and Design Basis Accident Analysis' [3]. The ONR SAPs [16] 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. In the UK context, when the potential consequences of these BDBEs is more severe, ONR refers to these as 'Severe Accidents'. SAs are well defined in the SAPs [16] (paragraph 664) in terms of radiological consequences and societal risk. As stated in the TAG NS-TAST-GD-007 guide on SAA [17], 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 [15].

15.2.2 Accident Management and Emergency Preparedness

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. 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.

EP is defined in the IAEA Safety Glossary [18] 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) apply 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 PSR Deliverables

The following deliverables will be produced and reported at PSR Revision 1.

15.2.3.1 Severe Accidents Gap Analysis Report

A gap analysis report will be produced to demonstrate the level of potential additional work which is needed to support safety claims and meet UK regulations. It will contain design alignment to the SAPs [16] and TAGs as well as outcomes of the safety demonstration approach.

The scope of the report will cover gaps which are identified following the technical guidance on GDA 'ONR-GDA-GD-007' [15] and technical guidance 'NS-TAST-GS-007' [17], to track the relevant requirements and expectations on SAA within the GDA cope.

15.2.3.2 Safety Concept for Severe Accidents Report

SAA is an area which demonstrates identification of challenges to the containment, ESFs dedicated and available to cope with these phenomena, and procedures and guidance implementing mitigation strategies. It is recognised the SMR-300 design informs many topics in safety substantiation which are input to SAA – Fault studies, Level 2 PSA, etc.

The aim of the report is to provide information about SMR-300 design specifics related to SA, phenomena applicable during SA progression as well as the technical means in place (SSCs and ESFs) to cope with them. This will be part of the overall demonstration of DiD and analysis for safety substantiation claims. At the PSR stage, the document will be based on identification of sequences and events leading to DEC for SMR-300. This will be done based on available PSA data and stand-alone safety justifications of ESFs and SSCs (Topic Engagement Plan (TEP) for Fault Studies). Requirements for analysis and phenomena assessment will be part of the report and will be supporting further PSR chapter development, including with items such as In Vessel Retention (IVR), Hydrogen, High Pressure Melt Ejection (HPME), etc including phenomena assessment. The report will contain responses to the gaps identified in the Gap Report. 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.

This deliverable provides the approach and methods to be used in order to substantiate the Claim 2.1.3.1 throughout to Claim 2.1.3.4.

15.2.3.3 Accident Management Programme

The Accident Management Programme report will be focused on providing guidance and basis for next step development up to the stage where site dependent conditions are considered. The report will contain SSCs which are credited with accident prevention, control and management with a link to their detailed description and safety and effectiveness demonstration. ESFs operation and effectiveness will be highlighted and linked to strategies and procedures which will be needed for a complete accident management plan. This information will be aligned with the Radiation Protection chapter in PSR Part B Chapter 10 ‘Radiological Protection’ [8]. In addition, some requirements and features of the emergency centre, control room, Post Accident Monitoring System, etc will be presented.

Accident management programme deliverable aims to demonstrate the approach to substantiate Claim 2.1.3.4 and with this to support higher level claim (2.1.3) justification.

15.2.3.4 ALARP Demonstration of Severe Accidents and Emergency Preparedness

The deliverable will provide a summary of the SAA topic, systemise gaps (from the Severe Accidents Gap Analysis Report), resolutions (from Safety Concept for Severe Accidents Report) and their implementation (given in the Accident Management Programme Report). The level of details will depend on the design advancement. Nevertheless, the main goal is a structured demonstration of SMR-300 design and its capability to meet regulatory expectation and explicitly demonstrate the design is ALARP from SA perspective (incl. challenges, means to cope with, effectiveness and consequences).

This deliverable is focused on deriving available information and guidance of methods and approach presented in the chapter and other deliverable, to help constructing the logic for design safety demonstration and respectively Claim 2.1.3.

15.3 BDBA, SAA AND EP CLAIMS, ARGUMENTS AND EVIDENCE

The primary purpose of a CAE approach is to capture the golden thread of a safety case narrative to demonstrate how plant and operational evidence is brought together and to justify that a high-level or fundamental claim is true. In the context of the GDA of the Generic SMR-300, that is how the Fundamental Purpose of the overarching SSEC (presented in PSR Part A Chapter 1 'Introduction' [1]) is achieved.

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 'Claims, Arguments & Evidence' [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.

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 nuclear safety assessment disciplines which are responsible for development of the nuclear safety assessment. This chapter presents the BDBA, SAA and EP topics for the Generic SMR-300 to support 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).

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

This Chapter is predominantly focused around three main areas:

- Deriving safety features to be placed on 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 to be adopted for developing accident management and emergency preparedness arrangements.

Claim 2.1.3 is decomposed into four further sub-claims which support these three main areas.

Claim 2.1.3.1 and Claim 2.1.3.2 presents 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, noting that the maturity of evidence for this claim will be limited at a PSR stage.

Claim 2.1.3.3 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 2.1.3.4 covers accident management and emergency preparedness.

Table 2 shows in which section of this PSR chapter these claims are demonstrated to be met.

Table 2: CAE Chapters

Claim No	Claim	Chapter Section
2.1.3.1	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.	15.5 BEYOND DESIGN BASIS ACCIDENTS
2.1.3.2	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.	15.6 SEVERE ACCIDENT ANALYSIS
2.1.3.3	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.	15.7 ENGINEERED SAFETY FEATURES
2.1.3.4	Accident management and emergency preparedness take all reasonably practicable measures to prepare for possible accidents, and to mitigate their consequences should they occur	15.8 ACCIDENT MANAGEMENT AND EMERGENCY PREPAREDNESS

A summary of the current CAE route map for PSR Part B Chapter 15 is provided in Appendix A which is taken from the Generic SMR-300 Overarching SSEC Claim Routemap presented in Appendix A of PSR Part A Chapter 3 [2]. A further update on claim decomposition, argument development and evidence maturity will be provided in the subsequent update of the chapter.

15.4 BEYOND DESIGN BASIS, SEVERE ACCIDENT AND ACCIDENT MANAGEMENT CODES, STANDARDS AND METHODOLOGIES

This subchapter outlines the code and standards related to BDBA, SAA and AM and EP used in the US and UK, together with applicable regulations and guidance.

15.4.1 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.

US NRC 10 CFR 50.155 Mitigation of beyond-design-basis events [19]

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).

HPP-160-3018, Design Standard for Severe Accident Design and Analysis Strategy [20]**.**

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 associated with CNSC and US NRC Regulatory Documents and provides acceptance criteria to ensure the requirements are met. Additional documentation, including IAEA documents that contain useful information pertinent to SAs design are also identified as guidance to designers and safety analysts.

The other US guidance related to these topics is specified in Table 3.

Table 3: US Guidance

Label	Title	Revisions
NRC RG 1.206	Combined License Applications for Nuclear Power Plants (PWR Edition) [21]	June 2007
NUREG-0800, Chapter 19	Standard Review Plan – Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors [22]	Rev. 3, December 2015
-	Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants [23]	August 1985
-	Policy Statement on Safety Goals for the Operations of Nuclear Power Plants [24]	August 1986
-	Policy Statement on Nuclear Power Plant Standardization [25]	September 1987
-	Policy Statement on Regulation of Advanced Nuclear Power Plants [26]	July 1994
-	The Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities [27]	August 1995
10 CFR PART 52	Licenses, Certification, and Approvals for Nuclear Power Plants. [28]	January 2022
SECY-90-016	Evolutionary Light-Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements [29]	January 1990

Label	Title	Revisions
SECY-93-087	Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs [30]	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 [31]	February 2009
NUREG/CR-2300, Vol. 1, Chapter 7	PRA Procedures Guide [32]	January 1983
NUREG/CR-6595	An Approach for Estimating the Frequencies of Various Containment Failure Modes and Bypass Events [33]	Rev. 1, October 2004

15.4.2 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 4.

Table 4: UK Guidance and RGP

Label	Title	Revisions
SAPs	Safety Assessment Principles [16]	1
ONR-GDA-GD-006	Generic Design Assessment Guidance to Requesting Parties [34]	Rev. 0
ONR-GDA-GD-007	Generic Design Assessment Technical Guidance [15]	Rev. 0
NS-TAST-GD-005	Guidance on the Demonstration of ALARP [35]	June 2023
NS-TAST-GD-006	Design Basis Analysis [36]	December 2022
NS-TAST-GD-007	Severe Accident Analysis [17]	December 2022
IAEA SSR-2/1	Safety of Nuclear Power Plants: Design [37]	Rev. 1
IAEA SSG-2	Deterministic Safety Analysis for Nuclear Power Plants [38]	Rev. 1
IAEA SSG-4	SSG-4: Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants [39]	
WENRA RHWG	Safety of new NPP designs [40]	March 2013

15.4.3 Codes and Standards for Accident Management and Emergency Preparedness

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.

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 PSAR Framework [41].

EP is defined in the IAEA Safety Glossary [18] 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 DiD apply 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.

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 [42] and 10 CFR Part 50 Appendix E, Emergency Planning and Preparedness for Production and Utilization Facilities [43]. 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 5.

Table 5: Accident Management and Emergency Preparedness

Label	Title	Revisions
IAEA GSR Part 7	Preparedness and Response for a Nuclear or Radiological Emergency [44].	2015
IAEA GS-G-2.13	Arrangements for Preparedness for a Nuclear or Radiological Emergency [45].	2007
IAEA GSG-1	Criteria for Use in Preparedness and Response for a Nuclear or Radiological Emergency [46].	2011
IAEA SSG-54	Accident Management Programmes for Nuclear Power Plants [47].	2019
IAEA TECDOC-953	Method for the Development of Emergency Response Preparedness for Nuclear or Radiological Accidents [48].	1997
IAEA	The Fukushima Daiichi Accident, Non-serial Publications, Technical Volume 3/5 [49].	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 [50].	2013
HM Government	How we regulate radiological and civil nuclear safety in the UK [51].	April 2021

In the UK, Section 5.7 of NS-TAST-GD-007 [17] 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) [52]] which determines the extent of the detailed and outline emergency planning zones. The Approved Code of Practice for REPPIR 2019 is presented in [52].

15.4.4 Lessons Learnt

To demonstrate understanding of ONR's expectations for the PSR, 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' [53], has been undertaken to make sure that RGP, Operating Experience (OPEX) and important lessons learnt are considered at this stage.

These lessons will be identified for PSR Revision 1.

15.5 BEYOND DESIGN BASIS ACCIDENTS

Claim 2.1.3.1: 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.

This subchapter outlines the approach to be adopted for the assessment of BDBAs, which include DEC-A events.

15.5.1 Evaluation of DEC-A Events

UK Regulatory expectations on BDBA are summarised in the technical guide ONR-GDA-GD-007 [15]. In this context, the BDBA are recognised as DEC-A conditions, and the two nomenclatures can be considered synonymous. SAs (DEC-B) are discussed in subchapter 15.6. The guide [15] 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 will be defined in the Safety Concept for Severe Accidents Report (to be issued at PSR Revision 1) (see 15.2.3.2). A summary of the overall approach is described below.

15.5.2 Identification of DEC-A events

As stated in the technical guide ONR-GDA-007 [15], events classification should not be limited to Design Basis Events (DBEs). Criteria need 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.

15.5.3 Analysis of DEC-A events

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 [38] 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.

15.5.4 Acceptance criteria

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.

15.5.5 CAE Summary

The approach to the identification and assessment of DEC-A events at PSR Revision 1 will be based upon available DBAA and PSA data and will be referenced in the deliverable Safety Concept for Severe Accidents Report.

15.6 SEVERE ACCIDENT ANALYSIS

Claim 2.1.3.2: 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.

This subchapter outlines the approach to be adopted for SAA, which include DEC-B events.

As stated in the SAPs [16], when an initiating event is highly unlikely and difficult to predict, it is not always reasonably practicable to incorporate the conservatively designed safety measures expected for design basis accidents. It is often 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 [16] base the definition of SA on numerical targets of radiological consequences. As stated in TAG NS-TAST-GD-007 [15], 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 [54] and WENRA [40] expectations for deterministic analysis of DEC associated with core damage (so called 'DEC-B' analysis). The objectives for the Generic SMR-300 as defined in TAG NS-TAST-GD-007 [15] 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.

This subchapter describes the approach adopted for SMR-160 in a US context.

This subchapter also addresses the approach to the evaluation of DEC-B Events in a UK context.

15.6.1 SMR-160 Severe Accident Analysis Strategy

HPP-160-3018, Design Standard for Severe Accident Design and Analysis Strategy [20] has been developed to provide a framework for SAA, based on the regulatory requirements specified in CNSC and US NRC Regulatory Documents. The overall approach to SAA is outlined below.

SAA starts with the accident sequences from the Level 1 PSA that result in core damage and other representative SA sequences. Transient analysis is run until the plant transitions into a controlled and stable state, ensuring all requirements and objectives are met. If a controlled and stable state cannot be reached or any requirement or objective is not met, the design is examined to determine whether additional design features should be added to the design, or if other external equipment is needed.

Deterministic safety assessments are performed for each of the bounding event sequences to determine accident progression and phenomenology, and to analyse the contribution of each representative accident sequence to the small and large release frequencies as part of the Level 2 PSA.

Sensitivity studies are performed to ensure there are no cliff edge effects in the analysis.

This approach will form the basis for DEC-B events for the Generic SMR-300 in the UK.

15.6.2 Evaluation of DEC-B Events in a UK context

The approach to the identification and assessment of DEC-B events for the Generic SMR-300 will be defined in the Safety Concept for Severe Accidents Report (to be issued at PSR Revision 1). 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 will be part of the overall demonstration of DiD and analysis for safety substantiation claims.

At the PSR stage, the document will be based on identification of sequences and events leading to DEC for SMR-300. This will be done 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 IVR, HPME, etc including phenomena assessment. The report will contain responses to the gaps identified in the Gap Report. 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.

The report will also consider lessons learnt from the Fukushima accident and how these have been applied to the design of the SMR-300.

A summary of the work undertaken in the Safety Concept for Severe Accidents Report will be provided at PSR Revision 1.

15.6.3 CAE Summary

The approach to the identification and assessment of DEC-B events at PSR Revision 1 will be based upon available PSA data and will be referenced in the Safety Concept for Severe Accidents Report deliverable.

15.7 ENGINEERED SAFETY FEATURES

Claim 2.1.3.3: 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.

The SMR-300 Generic Design has a variety of ESFs, strategies and procedures for responding to DBA, BDBA, and SAs. This subchapter 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 ‘Reactor Coolant System and Engineered Safety Features’ [6].

Based on the current SMR-160 design, a preliminary list of SSCs that are available for Severe Accident Mitigation is presented in Table 6, taken from HPP-160-3018 [20].

Table 6: 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 Ventilation System (CBV)	Remove heat from the containment to the Chilled Water System.
	Passive Containment Heat Removal System (PCHR)	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	Containment Ventilation System (CBV) purge function	Controlled depressurization 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.

Function	Systems/Components	Actions
	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 can be found in the SMR-160 Level 2 PSA [55].

15.7.1 CAE Summary

The SMR-300 SAA deliverables will provide reference to the ESFs required to prevent and mitigate SAs. The approach and the focus will be to demonstrate safety means are in place to cope with SA and to support plant safety.

Further information will be provided at PSR Revision 1.

15.8 ACCIDENT MANAGEMENT AND EMERGENCY PREPAREDNESS

Claim 2.1.3.4: Accident management and emergency preparedness take all reasonably practicable measures to prepare for possible accidents, and to mitigate their consequences should they occur.

This subchapter outlines the approach to be adopted for AM and EP.

15.8.1 Accident Management Programme

Although they are unlikely to be needed, severe accident management 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 errors and events initiated outside the plant. In general, severe accident management programmes 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 [16] 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 Nuclear Power Plant (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 emergency operating procedures 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 leading to a release of radioactivity to the environment. Even in these events the consequences can still be mitigated using available and, in some cases, dedicated plant equipment.

The Accident Management Programme report will be focused on providing guidance and basis for next step development up to the stage where site dependent conditions are considered. The report will contain SSCs which are credited with accident prevention, control and management with a link to their detailed description and safety and effectiveness demonstration. ESF operation and effectiveness will be 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 will be presented.

15.8.2 CAE Summary

The SMR-300 SAA will provide a frame for the EOIs and SAMGs required to be followed by the operators in order to prevent and mitigate SAs. This is to support the Claim 2.1.3.4

The AM and EP programme for the Generic SMR-300 will be shown to meet the requirements of NS-TAST-GD-007 [17] and REPPIR 2019 [52].

15.9 CHAPTER SUMMARY AND CONTRIBUTION TO ALARP

This subchapter provides an overall summary and conclusion of the BDBA, Severe Accidents Analysis and Emergency Preparedness Chapter and how this chapter contributes to the overall demonstration of ALARP for the Generic SMR-300. PSR Part A Chapter 5 'Summary of ALARP' [5] sets out the overall approach for demonstration of ALARP and how contributions from individual chapters are consolidated.

This subchapter therefore consists of the following elements:

- Technical Summary
- ALARP Summary
  - Review against Relevant RGP
  - Demonstration Against Risk Targets
    - Evaluation of Risk
  - Risk Reduction Options
  - GDA Commitments and Forward Actions (FAs)
- Conclusion

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

15.9.1 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 Pre-Construction Safety Report (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 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 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 that the combination of ESFs and EP plans can demonstrate that the consequences of SAs are minimised and shown ALARP.

15.9.2 ALARP Summary

15.9.2.1 Review against RGP

Due to the differing regulatory approaches between the UK and US, the DEC topic is assessed in differing ways. Within the UK, a DEC systematic analysis should be carried out in a way complementary to DBA and PSA. 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.

In the US a more prescriptive approach is taken regulated by NRC and DEC are analysed in the PSA domain. A further revision of this Chapter will include a more detailed review of the methodology against RGP.

15.9.2.2 Demonstration Against Risk Targets

The numerical targets against which the demonstration of ALARP is considered are provided in PSR Part A Chapter 2 [4]. ESFs will contribute to the demonstration of ALARP by comparison against the risk targets in two ways:

- By fulfilling safety functions under DEC conditions (e.g., shielding and containment) and thereby contributing to achieving Targets 1-3 [4].
- By contributing to the mitigation of consequences, in the case of DEC-B events, they will contribute to the achievement of accident risk, Targets 4-9 [4].

Risks above the Basic Safety Limit (BSL) are not acceptable. Risks below the BSL require a demonstration of ALARP proportionate to the level of risk. The Basic Safety Objective (BSO) represents the modern safety standards and expectations against which the UK Generic SMR-300 will be assessed.

15.9.2.2.1 Evaluation of Risk

At this time, the evaluation of the normal operations and accident risks against Targets 1-9 [4] has not been provided. This information will be presented in PSR Part B Chapter [8] for normal operations, and PSR Part B Chapter 14 ‘Design Basis Accident Analysis’ [9], PSR Part B Chapter 15 ‘Beyond Design Basis and Severe Accident Analysis, and Emergency Preparedness’ [56] PSR Part B Chapter 16 ‘Probabilistic Safety Analysis’ [10] for accident conditions.

15.9.2.3 Risk Reduction Options

This is a placeholder to identify and review relevant Position Papers and Design Decision Papers with a view to demonstrate which option(s) is/are ALARP.

It will summarise those option evaluations, and it will briefly explore if other risk reduction options have or could be considered and either:

- Present the ALARP argument for why those options have not been implemented.

- Present the ALARP argument for why those options will be implemented in future.
- Create a FA to consider the option(s) at some future point (noting this still must be a point where a meaningful design improvement could be made).

The process for the assessment of risk reduction options is presented in HPP-3295-0017, 'Generic Design Assessment Reference Design Process and GDA Prospective Design Change Register' [57]. PSR Part A Chapter 5 'ALARP Summary' [5] considers the holistic risk-reduction process for the Generic SMR-300.

15.9.2.4 GDA Commitments and Forward Actions

There are no GDA commitments identified for PSR Part B Chapter 15 'BDBA, Severe Accident Analysis, and Emergency Preparedness'.

FAs have been collated and are managed via the process described in PSR Part A Chapter 4 'Lifecycle Management of Safety and Quality Assurance' [58]. PSR Part A Chapter 5 'ALARP Summary' [5] describes the contribution of the forward actions to the ALARP argument.

15.9.3 Conclusions

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

- The Chapter Claims identified have been met to a maturity aligned with the current vision of the PSR. Further CAEs will be presented in due course as the design develops.
- The Severe Accidents Gap Analysis Report, to be produced for PSR Revision 1, will define the level of potential additional work which is needed to support safety claims and meet UK regulations.
- The Safety Concept for Severe Accidents Report, to be produced for PSR Revision 1, will define the approach to the identification and assessment of DEC events.
- The Accident Management Programme, to be produced for PSR Revision 1, will define the approach to AM and EP.
- The ALARP Demonstration of Severe Accidents and Emergency Preparedness, to be produced for PSR Revision 1, will provide a summary of the SAA topic, including gaps (from the Severe Accidents Gap Analysis Report), resolutions (from Safety Concept for Severe Accidents Report) and their implementation (given in the Accident Management Programme report).

Part A Chapter 5 of this PSR 'ALARP Summary' [5] concludes that it can be demonstrated that the Generic SMR-300 reduces risks to ALARP and that the Fundamental Purpose of the SSEC has been fulfilled.

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15.11 LIST OF APPENDICES

Appendix A BDBA, SAA and EP CAE Route Map

Table 7: Chapter B15 CAE Route Map

Overarching SSEC Claim	Chapter Claim/s	Chapter Sub-claim	Chapter Section:
Claim 2.1 – Nuclear Safety The nuclear safety assessment specifies the requirements for safety measures such that safety functions are fulfilled, informs operational and emergency arrangements and demonstrates that risk is tolerable and ALARP.	**Claim 2.1.3** Beyond design basis faults and severe accidents are appropriately identified and risk assessed to be tolerable As Low As Reasonably Practicable (ALARP)	**Sub-claim 2.1.3.1** 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.	**15.5** BEYOND DESIGN BASIS ACCIDENTS
		Sub-claim 2.1.3.2 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.	**15.6** SEVERE ACCIDENT ANALYSIS
		Sub-claim 2.1.3.3 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.	**15.7** ENGINEERED SAFETY FEATURES
		Sub-claim 2.1.3.4 Accident management and emergency preparedness take all reasonably practicable measures to prepare for possible accidents, and to mitigate their consequences should they occur.	**15.8** ACCIDENT MANAGEMENT AND EMERGENCY PREPAREDNESS