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0	First Issue to Regulators to support PSR v0
1	Second Issue to Regulators to support PSR v1

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Table of Contents

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List of Tables

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List of Figures

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16.1 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 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].

PSR Part B Chapter 16 of the PSR presents the Claims, Arguments and Evidence (CAE) for the Probabilistic Safety Assessment (PSA) for the SMR-300 and how PSA has been applied during the design development, noting the discussion regarding the SMR-160 PSA provided in sub-chapter 16.1.1.

16.1.1 Purpose and Scope

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

This chapter (Part B Chapter 16) links to the overarching claim through Claim 2.1:

Claim 2.1: 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.

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 PSA aspects for the generic SMR-300 and therefore directly supports Claim 2.1.4.

Claim 2.1.4: The Probabilistic Safety Assessment (PSA) demonstrates that the design of the generic Holtec SMR-300 is balanced such that risk is tolerable and As Low As Reasonably Practicable (ALARP).

As per New Nuclear Power Plants: GDA Technical Guidance [3], a full scope PSA is expected as part of the GDA submission. The SMR-300 PSA is currently under development by Holtec International. This PSA will be available, post Step 2 GDA, to support future licensing activities and it is acknowledged that a UK variant of the generic SMR-300 PSA will ultimately be produced. It is expected that this would be in support of a specific site licence application where the full scope of hazards can be characterised and assessed.

There is a complete Level 1 (L1) and Level 2 (L2) PSA for the SMR-160 design. These assessments cover internal reactor faults for At Power, and Low Power and Shutdown modes of operation. These PSAs do not provide assessment of fuel transfer route hazards and those

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associated with the Spent Fuel Pool (SFP). Furthermore, although assessment of External and Internal hazards has been undertaken for the SMR-160 design, this is not currently provided for consideration given the importance of SMR-300 design details (e.g. plant layout etc) and the site-specific aspects required to support these assessments.

Recognising the above, the strategy for this chapter at GDA Step 2 is to provide confidence that there are no significant fundamental shortfalls in the proposed PSA scope, methods, implementation and analysis. In doing so, this shall provide confidence that the design and operation of the SMR-300 will meet the Office for Nuclear Regulation (ONR) Numerical Targets and support demonstration that the associated risks are ALARP.

In support of this overall strategy, this chapter will draw on the findings of the SMR-160 L1 and L2 PSAs, supported by sensitivity studies aimed as estimating the likely risk delta associated with significant SMR-300 design changes. The evidence from these assessments will be presented against specific claims as required throughout the chapter.

The current PSA addresses radioactive releases from the reactor core as a result of internal events and hazards arising during at-power and shutdown modes of operation. It does not currently address radioactive releases from non-reactor core sources (e.g. the SFP), or internal and external hazards.

It is noted that a Level 3 (L3) analysis, used to evaluate offsite consequences at a potential site, has not been performed in GDA Step 2. However, this is not required to support the fundamental assessment of the design. Development of a Level 3 PSA is a key element of PSA development, post Step 2 GDA, as the design matures.

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 16.3.

The scope of this chapter covers the L1 and L2 PSA as set out in sub-chapter 16.2.

Sub-chapter 16.4 summarises the codes and standards used in the development of the SMR-160 L1 and L2 PSA, as well as giving a robust definition of UK PSA Relevant Good Practice (RGP) for the SMR-300 PSA post GDA Step 2.

Sub-chapter 16.5 describes how the PSA adequately reflects the generic SMR-300 design and intended operation. It includes a review of the differences between the SMR-160 and SMR-300, and the identification of corresponding sensitivity studies that have been undertaken during GDA Step 2.

Sub-chapter 16.6 covers the modes of operation and sources of radionuclide release that require PSA.

Sub-chapter 16.7 presents a review of the SMR-160 PSA to show that it has been developed using methods that are consistent with UK RGP

Sub-chapter 16.8 presents the PSA numerical results and risk insights for the internal events L1 and L2 PSA for both at power and shutdown conditions.

Finally, sub-chapter 16.9 provides a technical summary of how the overarching claim for PSA is met and a summary of the contribution from this chapter to support the demonstration that

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risks are likely to be tolerable and ALARP for the generic SMR-300, that is presented in PSR Part A Chapter 5 Summary of ALARP and SSEC [4]. Sub-chapter 16.9.3 also discusses any GDA Commitments that have arisen.

A GDA Commitment has been raised in Part B Chapter 14 Design Basis Analysis (Fault Studies) [5] (C_Faul_103) which reflects the requirement to provide a full scope PSA commensurate with UK RGP; with the target for resolution being issue of UK Pre-Construction SSEC (see also sub-chapter 16.9.3).

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 [6].

16.1.2 Assumptions

Assumptions which relate to this topic have been formally captured in the Commitments, Assumptions and Requirements process [7]. Further details of this process are provided in Part A Chapter 4 Lifecycle Management of Safety and Quality Assurance [8].

The identified over-arching assumption (A_PSA_042) is;

- Due to the design and operational uncertainties, the PSA level of detail is limited. Therefore, the PSA makes bounding assumptions to ensure that an appropriate safety margin exists with respect to any risk-informed information provided. These assumptions will need to be validated and have been summarised in the following identified PSA reports (see sub-chapter 16.5.2):
    - SMR-160 PSA Modelling Groundrules and Assumptions [9].
    - SMR-160 Low Power/Shutdown PSA [10].
    - SMR-160 L2 PSA [11].
    - PSA Based Seismic Margin Assessment for SMR-160 [12].
    - SMR-160 Fire PSA Quantification [13].
    - SMR-160 Internal Flood PSA [14].
    - SMR-160 External Flood Analysis [15].
    - SMR-160 High Winds PSA [16].

16.1.3 Interfaces with Other SSEC Chapters

The PSA chapter interfaces with the following PSR chapters.

Part A Chapter 2 General Design Aspects and Site Characteristics [6] describes the generic design aspects and site characteristics.

Part A Chapter 5 Summary of ALARP and SSEC [4] uses the output from the SMR-160 PSA and supplementary Sensitivity Analysis.

Part B Chapter 1 Reactor Coolant System and Engineered Safety Features [17] presents the design of Reactor Coolant System (RCS), reactivity control systems and associated systems modelled in the PSA (see sub-chapter 16.5.1).

Part B Chapter 2 Reactor [18] provides design information and radionuclide inventory information pertinent to the L2 analysis (see sub-chapter 16.6.3).

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Part B Chapter 4 Control and Instrumentation (I&C) [19], Part B Chapter 5 Reactor Support Systems [20] and Part B Chapter 6 Electrical Engineering [21] provide input into the detailed modelling within the PSA and are key to ensuring that the PSA adequately reflects the design of those systems, including failure modes, reliability data and assumptions (see sub-chapter 16.5.1).

Part B Chapter 10 Radiological Protection [22] utilises information provided by the PSA to evaluate post-accident conditions and operator dose exposures (see sub-chapter 16.5.1).

Part B Chapter 14 Design Basis Analysis (Fault Studies) [5] provides input to the assessment of the L1 PSA success criteria. The PSA and fault studies topic areas have been cooperating to provide confidence in the completeness of the list of initiating events identified for a facility (see sub-chapter 16.6.1). The Requesting Party's PSA modelling provides input to the Initiating Event Frequencies (IEF) identified in the fault schedule (see sub-chapter 16.8.3).

Part B Chapter 15 Beyond Design Basis Accident Analysis [23] provides the input to the assessment of the L2 PSA (see sub-chapter 16.6.3). An important area of interface is the identification of the scope of confirmatory analyses.

Part B Chapter 17 Human Factors [24] provides input to the assessment of the PSA's Human Reliability Analysis (HRA) (see sub-chapter 16.4.1 and 16.5.1). In addition, the PSA provides input to the identification of the human-based safety claims, Human Failure Events and evaluation of their importance to overall risk.

Part B Chapter 18 Structural Integrity [25] provides input to the assessment of Structural Integrity (SI), for example, the containment structural analysis (metallic parts) for the Level 2 PSA (see sub-chapter 16.5.1).

Part B Chapter 20 Civil Engineering [26] provides input to the assessment of the containment structural analysis for the L2 PSA and to the External Hazards PSA regarding definition of hazards' magnitudes and frequencies, and fragilities of structures (see sub-chapter 16.5.1).

Part B Chapter 21 External Hazards [27] and Part B Chapter 22 Internal Hazards [28] provide input to the review of the External and Internal Hazards prioritisation and PSA to ensure, for example, that assumptions in the model are aligned with the design and operational procedures and that the list of External and Internal Hazards considered is complete (see sub-chapter 16.6.1).

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16.2 OVERVIEW OF L1 & L2 PROBABILISTIC SAFETY ASSESSMENT

16.2.1 Level 1 PSA

The L1 PSA has been used to evaluate internal events and hazards for plant operating and shutdown modes. The purpose of the L1 PSA analysis is to evaluate the frequency and cause of accidents leading to core damage. The L1 PSA uses top-logic modelling based on the event tree method supported by a linked fault tree approach. The SMR-160 PSA methodology is described in detail in the following Holtec Reports:

- Level 1 PSA Methodology [29].
- Seismic PSA Methodologies [30] [31].
- Internal Fire PSA Methodologies [32] [33].
- Internal Flood PSA Methodology [34].
- Risk Significance Determination Methodology [35].
- SMR-160 Low Power/Shutdown PSA [10].
- SMR-160 External Flood Analysis [15].
- SMR-160 High Winds PSA [16].
- SMR-160 PSA Human Reliability Analysis [36].

While the Internal and External Hazards PSA modelling have been developed for the SMR-160, the relevance of the numerical results and specific insights to the SMR-300 design are less clear than for internal events due to significant changes in the control and reactor building configuration. As a consequence, only the methodologies for the SMR-160 hazards PSA evaluations (which will be replicated for the SMR-300) are presented in this PSR. Specific documents providing further details of the SMR-160 L1 PSA procedure, methodology and calculations are referenced in the text.

The numerical results and insights from the SMR-160 L1 and L2 internal events PSA are presented in this PSR, supported by the results from UK PSA Sensitivity Studies [37] These results are considered to representative of the SMR-300 (see Section 16.8.4) due to the similarity of the SMR-300 and SMR-160 accident mitigating system design. The SMR-160 results are provided in the following Holtec Reports:

- SMR-160 PSA Integration and Quantification Report [38].
- SMR-160 PSA Risk Significant SSC Determination for Internal Events [39].
- SMR-160 Low Power/Shutdown PSA [10].
- SMR-160 Level 2 PSA [11].

The documentation on the SMR-160 PSA was reviewed to determine the significance of any differences to be resolved during detailed design of the SMR-300 post GDA Step 2. The impact of the most significant differences were assessed in a set of sensitivity studies, referenced below; the results of which are used to support the discussion and conclusions presented in this chapter. These reviews are detailed in the following:

- Holtec SMR PSA Gap analysis between SMR-300 and SMR-160 [40].
- SMR-300 Internal and External Hazards Screening Review [41].
- PSA Review of Preliminary Fault Schedule [42].
- SMR 300 Passive Safety Systems Reliability Review [43].

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- Review of Current SMR-160 PSA against PSA best practice [44].
- UK PSA Sensitivity Studies [37].

16.2.2 Level 2 PSA

The L2 PSA evaluates the potential for radionuclide release external to the plant from a severe accident in a single unit. The L2 PSA is an extension of the L1 PSA and evaluates the progression of known core damage sequences, their impact on the containment system, and the potential release of radioactive materials into the surrounding environment. This analysis takes into account specific plant characteristics. By using the core damage sequences identified in L1, the L2 PSA calculates the probability and timing of various scenarios where the containment may fail, leading to the release of radioactive materials beyond the containment boundary. The SMR-160 L2 PSA methodology is described in detail in the Level 2 PSA Methodology [45] and the Risk Significance Determination Methodology [35].

The numerical results and insights from the SMR-160 L2 PSA [11] for a single unit site due to internal events are presented in this chapter and are judged to be broadly applicable to the SMR-300 (per unit of a twin unit site) due to the entirety of the L1 PSA being taken forward to the L2 analysis. Specific documents providing further details of the SMR-160 L2 PSA procedure, methodology and calculations are referenced in the text.

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16.3 PROBABILISTIC SAFETY ASSESSMENT CLAIMS, ARGUMENTS AND EVIDENCE

This chapter presents the PSA aspects for the generic SMR-300 and therefore directly supports Claim 2.1.4.

Claim 2.1.4: The Probabilistic Safety Assessment (PSA) demonstrates that the design of the generic Holtec SMR-300 is balanced such that risk is tolerable and As Low As Reasonably Practicable (ALARP).

Claim 2.1.4 has been further decomposed within Part B Chapter 16 to provide confidence that the relevant requirements placed on Structures, Systems, and Components (SSC) will be met during all lifecycle phases.

The PSA analysis is predominantly focused around three main areas:

- Providing an integrated, structured safety analysis that combines engineering and operational features in a consistent overall framework.
- Enabling complex interactions to be identified and examined and provides a logical basis for identifying any relative weaknesses and identifying appropriate and balanced mitigating features.
- Assessing the resulting risks from plant operations and in turn permitting comparison with numerical risk targets for workers and the public and the demonstration of ALARP.

This decomposition has been undertaken by breaking down Claim 2.1.4 into five further sub-claims which support these three main areas.

Claim 2.1.4.1 is an enabling claim for all of the above areas by ensuring that the characterisation and methodologies that the PSA will be assessed against are appropriate.

Claims 2.1.4.2 and 2.1.4.3 support the implementation of the PSA methodology to ensure that it has accurately reflected the design and operational features of the SMR-300 and that all significant sources of radioactivity, modes of operation and accident initiators have been identified and evaluated.

Claim 2.1.4.4 ensures that differences are identified between the United States (US) and UK regulatory approaches that the additional considerations that are necessary for the PSA model to be broadly aligned with UK PSA RGP have been met.

Claim 2.1.4.5 supports the appropriate use of the PSA in the design process (e.g. through system reliability analysis) and in evaluating the risk outcomes relative to numerical targets and ALARP.

Table 1 shows the breakdown of Claim 2.1.4 and identifies in which chapter of this PSR these claims are evidenced.

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Table 1: Claims Covered by PSR Part B Chapter 16

Claim No	Claim	Chapter Section
2.1.4.1	The scope and methodology of the PSA performed as part of the fault analysis and design development enables designers to achieve a balanced design, permit the interrogation of the risk profile in terms risk contributors accident release magnitude and frequency, and demonstrate numerical risk targets are met.	Sub-chapter 16.4 PSA Codes and Standards
2.1.4.2	The PSA adequately reflects the generic Holtec SMR-300 design and intended operation and in the absence of specific information assumptions are justified and their impact evaluated.	Sub-chapter 16.5 Differences Between the SMR-160 and the SMR-300
2.1.4.3	The PSA covers all significant sources of radioactivity, all permitted operating states and relevant initiating faults.	Sub-chapter 16.6 Modes of Operation and Sources of Radionuclide Release
2.1.4.4	The PSA model is broadly aligned with UK RGP in terms of level of detail, accident sequence development, dependencies, and success criteria.	Sub-chapter 16.7 PSA Review Against UK RGP
2.1.4.5	PSA has been used to inform the design processes, including modifications, risk, and ALARP outcomes, as well as supporting the Design Basis Assessment (DBA) and Beyond Design Basis Assessment (BDBA)	Sub-chapter 16.8 PSA Results and Insights

Appendix A provides a full Claims, Arguments and Evidence mapping for Part B Chapter 16, which includes any lower-level claims, arguments and evidence needed to support the claims in the table above. This includes identification of evidence available at PSR v1 and aspects for future development of evidence to support these claims beyond PSR v1.

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16.4 PSA CODES AND STANDARDS

Claim 2.1.4.1: The scope and methodology of the PSA performed as part of the fault analysis and design development enables designers to achieve a balanced risk profile, permit the interrogation of the risk profile in terms risk contributors accident release magnitude and frequency, and demonstrates numerical risk targets are met.

Claim 2.1.4.1 has been further decomposed into two arguments. US and International Regulatory and Industry policies, guidance and standards were utilised to perform the PSA and associated severe accident evaluations (A1) and, a robust definition of UK PSA RGP has been established to inform the development of an acceptable PSA for the SMR-300 post GDA Step 2 (A2).

Codes and standards that have been utilised to perform the PSA and associated severe accident evaluations are identified in detail in the Holtec Methodology Reports [29] [45] [30] [32] [34] [35] [31] [33]. The overarching codes and standards are listed in Table 2 below.

Table 2: Main Codes, Standards, and Regulations Utilised for Development of the PSA

Title of Code / Standard Reference	Rev / Date
10 Code of Federal Regulations (CFR) Part 50, Domestic Licensing of Production and Utilization Facilities [46].	January 1956
10 CFR Part 52, Licenses, Certifications, and Approvals for Nuclear Power Plants [47].	August 2007
Regulatory Guide (RG) 1.200, An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities, Revision 2 [48].	March 2009
RG 1.206, Combined License Applications for Nuclear Power [49].	June 2007
DC/COL-ISG-028, Assessing the Technical Adequacy of the Advanced Light-Water Reactor Probabilistic Risk Assessment for the Design Certification Application and Combined License Application [50].	November 2016
NUREG-0800, Standard Review Plan – Regulatory Treatment of Nonsafety Systems for Passive Advanced Light Water Reactors [51].	June 2014
SECY-13-0029, History of the Use and Consideration of the Large Release Frequency Metric by the U.S. Nuclear Regulatory Commission (U.S.NRC) RG [52].	March 2013
SSG-3, Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants [53].	April 2010
SSG-4, Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants [54]	May 2010
REGDOC 2.4.2, Probabilistic Safety Assessment (PSA) for Nuclear Power Plants [55].	May 2014
N290.17-16, Probabilistic safety assessment for nuclear power plants [56].	December 2016
Americal Society of Mechanical Engineers (ASME)/Americal Nuclear Society (ANS) RA-Sb-2013, Standard for L1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications [57].	September 2013
ANS/ASME-58.22-2014, L1 PSA Standard and the draft Low Power and Shutdown (LPSD) PSA Standard [58].	2014
ASME/ANS RA-S-1.2-2019, Severe Accident Progression and Radiological Release (Level 2) Probabilistic Risk Assessment (PRA) Standard for Nuclear Power Plant Applications for Light Water Reactors (LWRs) [59].	2019
NEA/CSNI/R(2009)16, Probabilistic Risk Criteria and Safety Goals [60].	December 2009
NUREG/CR-6823 Handbook of Parameter Estimation for Probabilistic Risk Assessment [61].	September 2003
NUREG-0492, Fault Tree Handbook [62].	January 1981
NUREG/CR-5485, Guidelines on Modelling Common-Cause Failures (CCFs) in Probabilistic Risk Assessment [63].	June 1998
Electric Power Research Institute (EPRI) NP-3583, Systematic Human Action Reliability Procedure (SHARP) [64].	June 1984

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Title of Code / Standard Reference	Rev / Date
NUREG-2199, An Integrated Human Event Analysis System (IDHEAS) for Nuclear Power Plant Internal Events At Power Application [65].	March 2017
NUREG/CR-6883, The SPAR-H Human Reliability Analysis Method [66].	August 2005

Table 3 summarises the applicable codes and standards that Holtec Britain considers to be relevant to L1 and L2 UK PSA RGP [44].

Table 3: Identified Applicable Codes and Standards Relevant to UK PSA RGP

Title of Code / Standard Reference	Rev / Date
SSG-3, Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants [67]	March 2024
SSG-4, Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants [54]	May 2010
ASME/ANS RA-S-1.1–2022, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications [68]	May 2022
ANS/ASME-58.22-2014, L1 PSA Standard and the draft LPSD PSA Standard [58].	2014
ASME/ANS RA-S-1.2-2019, Severe Accident Progression and Radiological Release (Level 2) PRA Standard for Nuclear Power Plant Applications for Light Water Reactors (LWRs) [59].	2019
GDA-GD-006, New Nuclear Power Plants: Generic Design Assessment Guidance to Requesting Parties [69].	October 2019
GDA-GD-007, New Nuclear Power Plants: Generic Design Assessment Technical Guidance [70].	May 2019
ONR CM9 Ref 2019/367414, Safety Assessment Principles for Nuclear Facilities [71].	January 2020
NS-TAST-GD-030, Nuclear Safety Technical Assessment Guide: Probabilistic Safety Assessment [72].	June 2019
NS-TAST-GD-063, Safety Technical Assessment Guide: Human Reliability Assessment [73].	December 2022
NS-TAST-GD-046, Computer Based Safety Systems [74].	December 2023

16.4.1 Codes, Policies, Guidance and Standards used in Developing the PSA

Argument 2.1.4.1-A1: US and International Regulatory and Industry policies, guidance and standards were utilised to perform the PSA and associated severe accident evaluations.

Evidence for Argument 2.1.4.1–A1:

L1 PSA Methodology [29]

The SMR-160 L1 PSA Methodology summarises the regulatory requirements, standards and industry best practices used in developing the methodology for performing a L1 PSA for internal events, at power and in shutdown modes of operation.

L2 PSA Methodology [45]

Regulatory requirements, standards and industry best practices used in developing the SMR-160 L2 PSA Methodology are referenced where applicable.

Seismic PSA Methodology [30] [31]

Provides the US regulatory and international requirements associated with performing a PRA-based seismic margin analysis: seismic fragility analysis and seismic plant response analysis. A SMR-160 methodology [30] is presented, and a subsequent SMR-300

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methodology [31] which is an update to the SMR-160 approach and draws on lessons learned from application of that method.

Internal Fire PSA Methodology [32] [33]

Lists the regulatory requirements, standards, and industry best practices used to determine the general approach to be followed to identify, define and quantify internal fire accident sequences. An SMR-160 methodology [32] is presented and a subsequent SMR-300 methodology [33] which is an update to the SMR-160 approach and draws on lessons learned from application of that method.

Internal Flood PSA Methodology [34]

Lists the regulatory requirements, standards, and industry best practices used to determine the general approach to be followed to identify, define and quantify internal flood accident sequences.

Risk Significance Determination Methodology [35]

Describes how the SMR-160 approach for determining SSC risk significance can be implemented in a manner consistent with meeting the safety goals and guidelines from regulatory requirements, standards, and industry best practices.

SMR-160 PSA Human Reliability Analysis [36]

Details the methodology used for determining the probabilities for pre and post-initiator human failure events modelled in the SMR-160 PSA.

Relevant US and international standards regulatory requirements, standards, and industry best practices were identified and shown to have been met in the development of the PSA methodologies defined above. When requirements differ, it has been demonstrated that the more restrictive of the requirements is used when developing the PSA.

A UK gap highlighted within each of the methodologies is that there has not been the incorporation of sequences for peer review, as may be required by some regulatory jurisdictions. In lieu of such or other jurisdiction specific quality constraints, independent review and approval practices under the performer’s quality program are required to be followed.

The PSA methodologies do not fully align with the expectations of UK RGP, including the requirement for Independent Peer Review (IPR) of the developed PSA models. Therefore, a Commitment (C_PSA_123) is raised to address these shortfalls for a SMR-300 PSA developed in support of a UK licence application. The Target for Resolution is issue of UK Pre-Construction SSEC (also see sub-chapter 16.9.3).

It is also noted that the HRA methodology is in adherence to NUREG-6883 but only in partial adherence to ONR Technical Assessment Guide (TAG) 063 [73]. As such a Commitment (C_Huma_003) has been raised in Part B Chapter 17 to close the identified gaps.

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16.4.2 UK Standards and Guidance for Future Benchmarking

Argument 2.1.4.1-A2: A robust definition of UK PSA RGP has been established to inform the development of an acceptable PSA for the SMR-300 post GDA Step 2.

Evidence for Argument 2.1.4.1–A2:

Review of Current SMR-160 PSA against PSA Best Practice [44]

Provides a high-level summary of UK PSA RGP in the context of additional considerations to ASME/ANS RA-S-1.1–2022 [1] and US regulatory guidance using relevant ONR TAGs and International Atomic Energy Agency (IAEA) guidance.

As nuclear power plants in the US were mostly built before the widespread adoption of PSA/PRA, U.S.NRC RG 1.200 [48] endorses ASME/ANS RA-Sa-2009 [75] (a previous revision of ASME/ANS RA-S-1.1–2022 [68]) as providing guidance to build a PRA of sufficient quality to be used for risk informed activities at plant (modifications, management of testing and inspections etc) with staff exceptions and clarifications. As RG 1.200 [48] was developed for currently operating reactors, U.S.NRC issued DC/COL-ISG-028 [50] for PSAs developed for the design/pre-construction application stages. DC/COL-ISG-028 [50] identifies supporting requirements in ASME/ANS RA-Sa-2009 [75] which are not applicable for Design Certification and Combined License (COL) applications, or cannot be achieved as written, as well as providing clarification to understanding how some of the other supporting requirements can be achieved.

A robust definition of UK PSA RGP [44] has been established to inform the development of a UK SMR-300 PSA to support any future licensing activities, as reflected in the endorsed PSA Design Challenge, HI-2241592 [76]. This PSA will be a derivative of the US SMR-300 PSA and be modified to align with the identified UK RGP and address any site-specific considerations.

Due to the differences between the US and UK regulatory approaches (e.g. the U.S.NRC not requiring PSA for fuel route or SFP) a high-level review was undertaken to identify the additional considerations to ASME/ANS RA-S-1.1–2022 [68] that are necessary for meeting UK PSA RGP. This review is detailed in the Review of Current SMR-160 PSA against PSA best practice [44], which uses the UK PSA RGP set out in relevant ONR PSA TAGs and IAEA guidance. Overall, the review of the documentation on the SMR-160 Level 1 At-Power Internal Events PSA can be considered broadly aligned with international standards and with ONR expectations. However, differences have been identified which will inform the update of the SMR-300 UK PSA, post GDA Step 2 and these include the following, as discussed elsewhere in this chapter:

- The use of minimum probabilistic limits set by the UK ONR for computer-based safety systems and HEPs, which are not mandatory but are strongly recommended.
- The completeness and bounding of IEs.
- The approach to Human Error identification and screening.

These issues will be addressed, post Step 2 GDA via work undertaken to address Commitment C_Faul_103 raised in Part B Chapter 14 (see sub-chapter 16.9.3).

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16.4.3 CAE Summary

The SMR-160 methodology for PSA has been undertaken based on the identified relevant US and international standards, regulatory requirements, and industry best practices. This approach has considered any differences between these requirements and adopted those deemed to be the most restrictive in the development of the PSA. However, there has not been the incorporation of sequences for peer review, as may be required by some regulatory jurisdictions.

Differences between the US and UK regulatory approaches (e.g. the U.S.NRC not requiring PSA for fuel route or SFP) have been identified, and the additional considerations necessary for meeting UK PSA RGP set out in relevant ONR PSA TAGs and IAEA guidance have been identified.

Whilst the methods adopted can be considered broadly aligned with international standards and with ONR expectations, the identified differences will need to be addressed in any update of the SMR-300 UK PSA, post GDA Step 2. This requirement is captured within Commitment C_Faul_103 (see Part B Chapter 14).

Furthermore, Commitment C_PSA_123 has been raised regarding the future need for IPR of the PSA.

Claim 2.1.4.1 from a methodology perspective, for a fundamental assessment, has been met to a maturity aligned with the current version of the PSR, noting the Commitments to close gaps against UK RGP (C_Faul_103 and C_PSA_123).

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16.5 DIFFERENCES BETWEEN THE SMR-160 AND THE SMR-300

Claim 2.1.4.2: The PSA adequately reflects the generic Holtec SMR-300 design and intended operation and in the absence of specific information assumptions are justified and their impact evaluated.

Claim 2.1.4.2 has been further decomposed into four arguments. Best-estimate methods, models and data are used where possible, otherwise conservative assumptions are made noting their impact on the PSA (A1). Due regard is given to the uncertainties in input probability and frequency values (A2) and the computer codes used have been qualified for the SMR-300 design (A3). Sensitivity studies have been performed to address any differences that impact SMR-300 PSA (A4).

16.5.1 Best-estimate Methods, Models and Data

Argument 2.1.4.2-A1: Best-estimate methods, models and data are used as far as possible within the PSA. Where this is not practicable, conservative assumptions have been made, justified and careful consideration taken of their impact on the analysis.

Evidence for Argument 2.1.4.2–A1:

SMR-160 PSA Data Selection Report Including CCF Parameters and Testing and Maintenance (T&M) Unavailabilities [77]

Documents the development of the SMR-160 PSA component database of failure rates, failure probabilities on demand for assessed component failures and data for CCF events.

SMR-160 PSA Modelling Groundrules and Assumptions [9]

Gives the groundrules and assumptions as a framework for the development of the SMR-160 PSA models to ensure completeness and consistency of modelling practices between analysts.

SMR-160 PSA Integration and Quantification Report [38]

Summarises the input data, methods, computer codes and assumptions for the L1 Internal Events At-Power SMR-160 PSA from the Accident Sequence Analysis, Success Criteria Analysis and System Level Notebooks.

SMR-160 L2 PSA [11]

Lists the principal inputs from the L1 PSA model and inputs specific to the L2 PSA modelling. Also gives the methods, computer codes and assumptions used in the development of the representative Core Damage Sequences, Containment Event Tree (CET) and L2 PSA modelling.

SMR-160 Low Power / Shutdown PSA [10]

Provides and categorises all the assumptions based on the potential impact to the L1 and L2 LPSD Internal Events PSA model and results. Summarises the PSA specific and SMR-160

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design specific input data, methods and computer codes used for the SMR-160 LPSD model.

PSA Based Seismic Margin Assessment (SMA) for SMR-160 [12]

Summarises the input data from the L1 and L2 PSA based models and seismic time history analyses for the design curves. Lists the methods, computer codes, assumptions and modelling notes that underlie the SMA.

SMR-160 Fire PSA Quantification [13]

Reference is made to the assumptions and input data from the preceding tasks of the Fire PSA. Assumptions that are associated with design changes and operational controls and constraints are provided in a consolidated list. Methodologies and computer codes are described for the different aspects of the Fire PSA.

SMR-160 Internal Flood PSA [14]

Lists the general assumptions that form the basis of the Internal Flood PSA and specific assumptions formulated in relevant section of the report. References all the principal inputs, methods and computer codes used to generate the Internal Flood PSA.

SMR-160 External Flood Analysis [15]

PSA specific and SMR-160 design specific input data, methods and computer codes are descried in the report. Details the modelling assumptions and notes specific to the External Flood PSA task.

SMR-160 High Winds PSA [16]

Summarises the modelling assumptions and notes specific to the High Winds PSA. PSA specific and SMR-160 design specific input data, methods and computer codes are described in the report.

Each of the Holtec PSA Reports demonstrate that best-estimate methods, models, computer codes and data are used where possible. The detail of the PSA is sufficient to meet the U.S.NRC guidance in DC/COL-ISG-028 [50]. However, due to the design and operational uncertainties at this stage of the design process the level of detail is limited. Therefore, the PSA makes bounding assumptions to ensure that an appropriate safety margin exists with respect to risk-informed information provided by the PSA. A consolidated list of assumptions has been summarised in the above mentioned reports, which will need to be validated as part of the ongoing process of PSA development post Step 2 GDA.

To be effective in supporting the design process and to provide meaningful results with regard to judging the overall risk posed by the design, the PSA reflects a level of detail limited only by the following:

- The availability of certain design details, operating procedures, and other information.
- The level at which useful reliability data are available.

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At the present time, elements of the detailed design that are not available to support the PSA include the following:

- The specific layout and location of equipment and cabling.
- The full and accurate capability of equipment and equipment operating characteristics.
- Plant-specific and operating data and procedures.
- Plant-specific experience to support HRA.
- Plant walkdowns cannot be performed to gain as-built insights.
- Plant-specific maintenance and testing schedules or data are unavailable.
- There are no similarly designed plants for comparison.
- A site has not been selected to support identification and evaluation of site-specific External Hazards.

Analysis has been performed that is consistent with the level of detail available and assumptions made have been refined, within the context of the available information, to avoid masking risk contributors from other sources due to overly conservative treatment.

Further details on the input data for the SMR-160 PSA are provided in the other relevant PSR chapters identified in sub-section 16.1.3 – 'Interfaces with Other SSEC Chapters'.

16.5.2 Uncertainty

Argument 2.1.4.2-A2: Due regard is given to the uncertainties in input probability and frequency values used, and their impact on the results.

Evidence for Argument 2.1.4.2–A2:

SMR-160 PSA Data Selection Report Including CCF Parameters and T&M Unavailabilities [77]

Presents the SMR-160 Component Reliability Database which includes the distribution type (beta, gamma or lognormal) and the error factor or variance for all component failures.

SMR-160 PSA Sensitivity and Uncertainty Analysis [78]

Provides the aleatory uncertainty distribution for the calculated Core Damage Frequency (CDF) and the epistemic uncertainty is identified and the impact on the PSA model assessed. The sensitivity of the results to SMR-160 PSA model assumptions in seven different cases is included.

SMR-160 L2 PSA [11]

Epistemic uncertainty issues from the L1 analysis are verified, additional L2 issues identified and the impact on the PSA model assessed. Aleatory uncertainty is defined using the UNCERT software to derive statistical uncertainty distributions about Large Release Frequency (LRF). Sensitivity analyses of qualitative uncertainty and plant specific considerations are included.

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SMR-160 Low Power/Shutdown PSA [10]

Key epistemic uncertainty issues associated with the LPSD PSA is identified and the impact on the PSA model assessed. Aleatory uncertainty is estimated using the UNCERT software to derive statistical uncertainty distributions about CDF. Sensitivity studies are performed for those qualitative areas and plant specific considerations identified as requiring further insights.

As per NUREG-1855 [79], uncertainty has been classified into two distinct but equally important contributions in the PSA for risk informed decision making. These are aleatory uncertainty and epistemic uncertainty.

Aleatory uncertainty is due to the randomness of the nature of events or phenomena. For PSA the most discussed uncertainty contribution involves the random occurrence of postulated events. These fall under the category of aleatory uncertainty since there is a range of possible values based on statistics and the value may vary over a range of potential numeric values based on a convolution of the individually generated uncertainties for inputs to the model. Examples of aleatory uncertainty include IEFs, equipment failure probabilities, and human failure event probabilities. Uncertainties introduced in this manner were most often addressed with a statistical distribution assigned to the input and then assembled into a combined uncertainty by using a sampling technique such as Monte-Carlo modelling or Latin-Hypercube to propagate the uncertainties and create a probability distribution for the CDF and LRF results.

Epistemic uncertainty is associated with the lack of knowledge about an event, system, phenomena, or model. The assessment of epistemic uncertainty has required more of a qualitative assessment of confidence in the generated results based on an understanding of the PSA inputs, assumptions and models. Three types of epistemic uncertainty were considered, Parameter, Model and Completeness. The impact on the PSA is assessed in accordance with EPRI 1016737 [80] to determine whether the uncertainty is judged to be of sufficient significance that it is a candidate for sensitivity assessment.

Uncertainty parameters were developed for each input to the PSA model. The uncertainty parameters were added into the database file associated with the PSA model for internal events with the unit at full power operation. The CDF or LRF cut sets were then used along with the PSA model database file to define the associated aleatory uncertainty characteristics using the UNCERT software, which is part of the EPRI developed Computer Aided Fault Tree Analysis (CAFTA) suite described in sub-chapter 16.5.3, to derive statistical uncertainty distributions for CDF or LRF.

Sensitivity analyses were conducted to evaluate the sensitivity of the PSA model in terms of CDF and LRF results to the uncertainties identified as requiring a sensitivity case to gain insights. In addition to these cases other plant-specific considerations of the SMR-160 were examined to determine the sensitivity of the results to identified significant design and operational considerations.

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16.5.3 Computer Codes

Argument 2.1.4.2-A3: The computer codes used for analysing the progression of severe accidents have been qualified for the design of the NPP under evaluation.

Evidence for Argument 2.1.4.2–A3:

CAFTA Software Verification and Validation [81]

This report has demonstrated the CAFTA computer code version 6.0b correctly calculates basic event failure probabilities, correctly identifies cut sets based on user input fault tree logic and correctly calculates individual cut set probabilities and overall top gate probabilities.

SMR-160 PSA MELCOR Model Verification Testing [82]

Demonstrates consistency of the MELCOR version 2.2 predictions to those of RELAP5 calculations without core damage and with expectations for core damage based on published studies considering differences in the SMR-160 design and other scaling considerations.

RELAP5-3D Verification and Validation Report [83]

Demonstrates how the RELAP5-3D version 4.4.2 has been verified using several generic tests cases to ensure the installation was performed correctly. A partial validation of RELAP5-3D version 4.4.2 the SMR-160 is provided.

SMR-300 RELAP5-3D Verification and Validation Plan [84]

Provides the verification and validation plan for performing the Evaluation Model Development and Assessment Process (EMDAP) for the RELAP5-3D computer code.

The PSA was created in the CAFTA computer code version 6.0b. As detailed in the CAFTA Software Verification and Validation [81], to verify the ability of the CAFTA software to generate accurate failure probabilities, a comparison of the computer-generated probabilities to hand calculations was performed. The CAFTA computer code has been shown to correctly calculate basic event failure probabilities using the various formulae options in the basic event database. The CAFTA code also correctly identifies cut sets based on user input fault tree logic and correctly calculates individual cut set probabilities and overall top gate probabilities.

The MELCOR computer code version 2.2 is used to evaluate the deterministic accident progression and models the severe accident phenomena and fission product behaviour. The results of the verification testing studies documented in the SMR-160 PSA MELCOR Model Verification Testing [82] demonstrate consistency of the MELCOR predictions to those of RELAP5 calculations for three representative cases not involving core damage. In addition, the results of MELCOR simulation of a representative accident scenario for SMR-160 that results in core damage are shown to be consistent with expectations based on published studies for traditional Pressurised Water Reactors (PWRs) and reflecting the differences in the design of SMR-160 and other scaling considerations.

The RELAP5-3D computer code is used for the simulation of light water reactor coolant systems during postulated accidents. The RELAP5-3D computer code version 4.4.2 was

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verified using several generic tests cases to ensure the installation was performed correctly. The validation of the RELAP5-3D computer code version 4.4.2 is incomplete and will need to be fully validated through an integral and separate effect test programme specific to the design. The verification and validation plan for the RELAP5-3D computer code [84] outlines the steps to establish the requirements of RELAP5-3D, develop an assessment base, develop the evaluation model, and assess the evaluation model, post GDA Step 2. The RELAP5-3D computer code will be validated by comparing computer code predictions to experimental data. Scaling analyses will be performed to ensure that the results are applicable to the prototypic SMR-300 design.

16.5.4 Review of Differences and Identified Sensitivity Studies

Argument 2.1.4.2-A4: The differences between the SMR-160 and SMR-300 which impact the PSA, have been clearly defined and sensitivity studies have been performed to address these differences.

Evidence for Argument 2.1.4.2–A4:

Holtec SMR PSA Gap analysis between SMR-300 and SMR-160 [40]

The SMR-160 is compared with the SMR-300 plant configuration and system designs to highlight the expected impact of the design differences on the SMR-300 PSA. Conclusions from this report are used to identify a series of SMR-160 sensitivity analyses.

UK PSA Sensitivity Studies [37]

Summarises the modelling changes and results of the sensitivity studies undertaken on the SMR-160 At Power PSA to capture expected design differences for SMR-300; and to address known differences in UK regulatory expectation.

SMR 300 Passive Safety Systems Reliability Review [43]

Summarises and reviews the US approach to evaluating passive safety system reliability assessment in the context of PSA against RGP, and provides recommendations for updates to the SMR-300 assessment of passive safety system reliability, post UK GDA Step 2.

While a change in Holtec SMR power levels results in a corresponding change in the specification of component duties and sizes, system flow rates, volumes, and water inventories etc, the overall design concept (types of system, number of trains, components, and associated success criteria for accident mitigation) generally remains unchanged. Likewise, the internal events accident sequence development and overall risk profile are expected likely to be similar, as discussed below.

In particular, passive safety system failures should not significantly influence the frequency of core damage or radiological release. In most cases, compared to electromechanical failures, the contribution of passive failures to the overall system failure probability is negligible, the exception being Secondary Decay Heat Removal System (SDH) (and the Primary Decay Heat Removal System (PDH) for the SMR-300), where the failure of passive elements due to accumulation of non-condensable gases are considered. The contribution from these failures have been assessed in the Sensitivty Studies PSA [37] and shown to be unimportant with respect to their contribution to core damage.

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However, specific differences between the SMR-160 and SMR-300 that impact the PSA have been identified and been included in the set of SMR-160 sensitivity studies performed during Step 2 of GDA [37]. In addition, the output from the reviews of the PSA identified a number of potential areas of divergence between UK and US PSA that were also addressed within the scope of the sensitivity studies. The following sensitivity studies were performed to address these key differences (the results of which are presented in sub-chapter 16.8.4):

- Changes to the PDH design from a two-loop system using the Annular Reservoir (AR) as heat sink to a one-loop system using the Passive Core Makeup Water Tank (PCMWT) as heat sink.
- Expected changes to the reliability claimed for the Plant Safety System (PSS) and Diverse Actuation System (DAS).
- Removal from the PSA of initiating events modelling a break in the upper riser and lower riser of the steam generator for SMR-160.
- Addition of a Large Loss of Coolant Accident (LLOCA) initiating event due to changes in the RCS.
- Addition of a Loss of Offsite Power (LOOP) initiating event as the assumption of a 100% successful operation on base load has been removed for SMR-300 at Palisades.
- Increase in the Vessel Loss of Coolant Accident (LOCA) initiating event frequency to align with UK expectations for failure of a Very High Reliability component such as a Pressure Vessel.
- Inclusion into the internal events PSA of the Feedwater Line Break (FWLB) and Main Steam Line Break (MSLB) Fault Trees modelling the breaks inside containment.
- Addition of the probability of failure of the passive systems for PDH and the SDH.

Additionally, the SMR 300 Passive Safety Systems Reliability Review [43] also included a review of the key assumptions, claims, and limitations of the Holtec International reliability approach to determine their reasonableness for consideration in the SMR-300 passive system reliability assessment post GDA Step 2. This review concluded that the assumptions/claims are generally reasonable for consideration in the SMR-300 passive system reliability assessment during GDA Step 2, however post GDA Step 2, these would still require further justification.

Recognising the claims-argument-evidence expectation in the UK, the assumptions used for the SMR-160 will be reviewed periodically as part of the SMR-300 passive system reliability assessment post GDA Step 2, to ensure these are up-to-date and if still relevant provide a more robust justification for each assumption or claim.

The risk significance for internal hazards is highly dependent on the layout of the buildings and degree of physical and spatial separation achieved between the source of the hazard (e.g. fire source) and vulnerable equipment and cable targets, as well as the separation between redundant equipment and targets. Since there have been significant changes in the Reactor Building and Control Building layouts, comparisons between the risk profiles associated with the internal hazards for the SMR-160 and SMR-300 designs cannot be conclusively drawn at the present time.

As part of the Commitment (C_Faul_103) to update the PSA to reflect the SMR-300 post GDA Step 2, all design and operational changes in moving from the SMR 160 to the SMR 300 will

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be considered. Target for Resolution is issue of UK Pre-Construction SSEC (see sub-chapter 16.9.3).

16.5.5 CAE Summary

Best-estimate methods, models, computer codes and data have been used to an extent consistent with the level of detail available. The PSA makes bounding assumptions to ensure that an appropriate safety margin exists with respect to risk-informed information provided by the PSA.

Uncertainty has been classified into aleatory uncertainty and epistemic uncertainty and the impact on the PSA assessed. Sensitivity analyses were conducted to evaluate the sensitivity of the PSA model in terms of CDF and LRF results to significant uncertainties and significant plant-specific considerations to gain insights.

Both the CAFTA software version 6.0b and MELCOR computer code version 2.2 were validated for L1 and L2 PSA modelling. There is a verification and validation plan for the RELAP5-3D computer code to assess the evaluation model post GDA Step 2.

The PSA directly relates to the existing SMR-160 design and will be updated to reflect the SMR-300 post GDA Step 2. The SMR-160 and SMR-300 designs are fundamentally similar in terms of accident prevention and mitigation. The results and insights are therefore considered to be representative for the internal events. This conclusion that the risk profile is broadly aligned is supported by a set of sensitivity studies which have been undertaken to assess the known significant design differences apparent at this stage of the design. Plant layout design differences impacting the risk from hazards will be evaluated at the site-specific design stage.

Claim 2.1.4.2, from a representation of the SMR-300 design perspective, has broadly been met to a maturity aligned with a fundamental assessment, for the current version of the PSR. Commitment C_Faul_103 is raised to address this.

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16.6 MODES OF OPERATION AND SOURCES OF RADIONUCLIDE RELEASE

Claim 2.1.4.3: The PSA covers all significant sources of radioactivity, all permitted operating states and relevant initiating faults.

Claim 2.1.4.3 has been further decomposed into four arguments. All potential initiating faults have been identified and the screening criteria used to exclude faults have been justified (A1). A strategy has been provided for consideration of additional faults (A2). A comprehensive set of severe accident phenomena has been evaluated and the L2 PSA is bounding (A3). The PSA addresses the risk associated with all permitted operating states (A4).

16.6.1 Initiating Faults in PSA

Argument 2.1.4.3-A1: All initiating faults have been identified that have the potential to lead to any person receiving a significant dose of radiation, or to a significant quantity of radioactive material escaping from its designated place of residence or confinement. Screening criteria used to exclude faults have been justified.

Evidence for Argument 2.1.4.3–A1:

Initiating Events Analysis for the SMR-160 PSA [85] [86]

Candidate events were identified from industry accepted standards and prior PSAs. The candidate initiating events were compared with the SMR-160 design, safety analysis information and operating characteristics to assess their specific applicability [85]. This document has been updated to assess the SMR-300 following the same approach [86].

SMR-160 Low Power/Shutdown PSA [10]

Initiating events are identified from the SMR-160 At-Power PSA and from failures of systems which ensure the safe state of the Plant within each Plant Operating State (POS).

SMR-160 PSA Screening of Other External Events [87]

Summarises the process and the results associated with identification and screening of internal and external hazards for modelling in the SMR-160 PSA modelling.

A thorough and systematic search was performed to define the spectrum of initiating events that could occur. This list of accidents includes both design basis events, as well as beyond design basis events. Potential initiating events (unscreened initiating events from various sources which have the potential of being an initiating event) were identified based on generic industry lists of initiating events that reflect industry experience with currently operating PWRs. These were supplemented with initiating events identified from a review of plant-specific system and design features (using Failure Modes and Effects Analysis (FMEA) and a Master Logic Diagram (MLD)) which considered potential for CCF system interfaces and spatial interactions.

The list of initiating events identified in the full power PSA were reviewed to determine if their occurrence would challenge a safety function for each POS. Additionally, the systems relied

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on to maintain the LPSD safety functions were reviewed for potential initiating events that are applicable to LPSD conditions but not applicable to at power operations.

The internal and external hazards that may affect the plant risk profile (e.g., earthquakes, floods, fires, tornadoes, aircraft impacts, and explosions) were identified based on a list of events created from past PSA studies, the PSA Procedures Guide NUREG 2300 [88] and the ASME/ANS Standard Ra-Sa-2013 [57]. Once the hazards were identified for consideration, the guidance in ASME/ANS Ra-Sa-2013 [57] was used to implement a progressive screening process to identify which hazards could be permanently screened from detailed evaluation and those that would require a quantitative hazard evaluation at the design stage or site-specific stage.

Notwithstanding the above, this process has not been supported by a systematic ‘bottom up’ hazard identification activity undertaken for the SMR-300 which would align with UK RGP. Furthermore, it is recognised that the bounding of IEs in the PSA does not currently align with accepted UK RGP.

Consequently, a Commitment, C_Faul_103, is raised (in Part B Chapter 14) to undertake, as necessary, supplemental safety activities to incorporate the full scope UK SMR-300 design. These will be targeted to ensure a holistic and comprehensive approach across the recognised safety assessment disciplines, including completion of the identification of Postulated Initiating Events (PIE) and the harmonisation between this initiating event list for use in both deterministic and probabilistic assessments. The target for resolution is issue of a UK Pre-Construction SSEC.

Further details of the input information used for the derivation of IEs are provided in the following chapters (discussed further in sub-chapter 16.1.3):

- Part B Chapter 14 Design Basis Analysis (Fault Studies) [5].
- Part B Chapter 21 External Hazards [27].
- Part B Chapter 22 Internal Hazards [28].

16.6.2 Consideration of Additional Faults

Argument 2.1.4.3-A2: A strategy has been provided for consideration of additional faults that have been identified as being within the scope of the PSA and justifications has been provided for any exclusions.

Evidence for Argument 2.1.4.3–A2:

PSA Review of Preliminary Fault Schedule [42]

Reviews the PIEs in Revision 0 of the Preliminary Fault Schedule (PFS) [89] and assesses them for inclusion in the UK PSA scope. Compares the IEFs claimed in the PFS [89] and the SMR-300 PSA IEFs and identifies any relevant differences to be considered post Step 2 GDA.

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SMR-300 PSA Screening of Other External Events [90]

The identification and screening of potential internal and external hazards for the SMR-160 PSA modelling is updated using ASME/ANS RA-S-1.1–2022 [68] for modelling in the SMR-300 PSA.

SMR-300 Internal and External Hazards Screening Review [41]

Provides Holtec Britain's plan on screening based on RGP for internal and external hazard screening in the UK. Reviews the hazard identification and screening by Holtec International [90] to identify the updates required to the SMR-300 PSA post GDA Step 2.

The PIEs in Revision 0 of the PFS are assessed [89] for inclusion in the UK PSA scope post GDA Step 2. A comparison between the PIEs in the PFS is made against the scope of existing Holtec International PSAs. These activities combined have allowed any apparent discrepancies in the scope of hazards identified in the PFS (and hence required to be included in a UK PSA) and those IEs assessed in the US derived PSA to be identified. Commentary is also provided on the intended approach to the treatment of some hazards in the UK PSA as appropriate. This is to provide stakeholders with insight into the approach to be adopted, and to identify if the treatment of the hazards in the UK PSA differs in any way to how it is presented in the PFS, or how it is currently addressed in the Holtec International PSAs. Additionally, the IEFs claimed in the PFS [89] were compared with the SMR-300 PSA IEFs, which identified any relevant differences to be considered post Step 2 GDA. However, as discussed in Sub-Section 16.6.1, Commitment C_Faul_103 is raised to ensure future alignment between PIEs assessed across all fault study areas as appropriate.

The potential internal and external hazards that may affect the plant risk profile for the SMR-160 in reference [81] were updated using the list from the ASME/ANS RA-S-1.1–2022 [68] for the SMR-300 PSA. Once the hazards were identified for consideration, the guidance in ASME/ANS RA-S-1.1–2022 [68] was used to implement a progressive screening process to identify which hazards could be permanently screened from detailed evaluation and those that would require a quantitative hazard evaluation at the design stage or site-specific stage.

The SMR-300 Internal and External Hazards Screening Review [41] summarises the relevant ONR guidance, standards and previous GDAs to provide Holtec Britain's plan on screening based on RGP for internal and external hazard screening in the UK, with a comparison against the US-based Holtec International approach. Against this background, the results of the identification and screening of internal and external hazards by Holtec International [90] have been compared against those derived by Holtec Britain to determine the additional hazards that will need to be included in the PSA for the SMR-300 UK GDA, as well as those that are provisionally screened out but will need to be revisited (e.g. once design is finalised or specific site is chosen). As part of the Commitment to update the PSA to reflect the SMR-300 post GDA Step 2 (see C_Faul_103), this will include the identified additional hazards in the PSA, as well as those deemed to require a revisit. Target for Resolution – Issue of UK Pre-Construction SSEC (see sub-chapter 16.9.3).

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16.6.3 Severe Accident Phenomena

Argument 2.1.4.3-A3: A comprehensive set of severe accident phenomena has been evaluated to determine their applicability and the L2 PSA is bounding in that it does not credit mitigating systems or capabilities that are relevant only to a radionuclide release.

Evidence for Argument 2.1.4.3–A3:

SMR-160 L2 PSA [11]

Provides the process and results of the identification of applicable severe accident phenomena (e.g. In-vessel / ex-vessel steam explosions, direct containment heating etc) and their potential impact to the plant SSCs.

It was assumed that all severe accident phenomena that are applicable to traditional PWRs, were also applicable to the SMR-160, unless a valid justification was provided to screen phenomena from the L2 PSA. The evaluation considered phenomena which have a potential to threaten the RCS pressure boundary and/or containment structural integrity listed in NUREG-1150 [91], NEA/CSNI/R(2009)5 [92], IAEA SSG-4 [54], IAEA Safety Reports Series No. 32 [93]. The assessment also identified and considered plant design aspects considered important to severe accidents (e.g. potential for containment bypass).

The L2 PSA is bounding in that it does not credit mitigating systems or capabilities that are relevant only to a radionuclide release. More specifically, mitigating factors such as fission product deposition, retention or scrubbing of fission products (e.g. filtration through the AR), or deflection or absorption are not credited in evaluating release source terms.

As part of the Commitment to update the PSA to reflect the SMR-300 post GDA Step 2 all sources of radionuclide release will be considered (see Commitment C_Faul_103, Part B Chapter 14). The target for Resolution is issue of UK Pre-Construction SSEC (see sub-chapter 16.9.3). Further discussion is provided in the following chapters (discussed further in sub-chapter 16.1.3):

- Part B Chapter 2 Reactor [18].
- Part B Chapter 15 Beyond Design Basis Accident Analysis [23].

16.6.4 Operating States

Argument 2.1.4.3-A4: The PSA addresses the risk associated with all permitted operating states.

Evidence for Argument 2.1.4.3–A4:

SMR-160 PSA Summary Report [94]

Provides a summary of the development and results of the Revision 0 SMR-160 L1 Internal Events At-Power PSA to calculate CDF.

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SMR-160 Low Power/Shutdown PSA [10]

Describes the L1 CDF and L2 LRF LPSD Internal Events PSA model developed for the SMR-160. This includes the identification of specific LPSD phases and their corresponding POS.

SMR-160 L2 PSA [11]

Details the development and results of the SMR-160 L2 Internal Events At-Power PSA that takes into account the core damage sequences from the L1 Internal Events PSA.

SMR-160 Internal Events PSAs have been conducted for both L1 CDF and L2 LRF at full power operations. The LPSD Internal Events PSA addresses risk associated with POSs other than full power operation for both L1 CDF and L2 LRF, including low power operation, refuelling outages, hot shutdown, and cold shutdown conditions. Plant-specific Refuelling Outage Milestones, associated activities and plant configurations were defined and grouped into a limited set of POSs.

The scope of the LPSD analysis excluded POSs that involved sources of radioactive material other than nuclear fuel within the reactor vessel, i.e., only core damage accidents were considered. Thus, accidents involving nuclear fuel in the SFP, in dry storage, or in transit were excluded from the scope.

As part of the Commitment to update the PSA (C_Faul_103) to reflect the SMR-300 post GDA Step 2 all modes of operation will be considered. Target for Resolution – Issue of UK Pre-Construction SSEC (see sub-chapter 16.9.3).

16.6.5 CAE Summary

Potential initiating events were identified based on generic industry and plant-specific system and design features. The list of initiating events identified in the L1 Internal Events At-Power PSA was reviewed to determine their applicability in the LPSD Internal Events At-Power PSA. Internal and external hazards that may affect the plant risk profile were identified from regulatory guidance, industry standards, and past PSA studies. Noting that these hazards are not quantified within the scope of the PSA for GDA Step 2.

However, it is recognised that supplementary analysis is required, post GDA Step 2, with regard HAZID for the SMR-300 to ensure UK RGP is met (C_Faul_103).

The PIEs and IEFs in the PFS [89] were reviewed and assessed for inclusion in the UK PSA scope post GDA Step 2. Holtec Britain has provided a plan on hazard screening based on RGP for internal and external hazard screening in the UK. Against this background, additional hazards were identified that will need to be included in the PSA for the SMR-300 UK GDA, as well as those that are provisionally screened out but will need to be revisited post GDA Step 2.

Severe accident phenomena which have a potential to threaten the RCS pressure boundary and/or containment structural integrity have been identified from regulatory and industry guidance. The L2 PSA is bounding in that it does not credit mitigating systems or capabilities that are relevant only to a radionuclide release. The Internal Events PSA addresses risks associated with full power operation and other POSs for both L1 CDF and L2 LRF, including

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low power operation, refuelling outages, hot shutdown, and cold shutdown conditions. It does not currently address all modes of operation (SFP, fuel transfer etc), and hence all sources of radionuclide release, within the facility.

The issues described in this summary have been captured and within Commitment C_Faul_103 (See Part B Chapter 14)

Claim 2.1.4.3 from a completeness of scope perspective, for a fundamental assessment, has not been met to a maturity aligned with the current version of the PSR and Commitment C_Faul_103 is raised to address this.

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16.7 PSA REVIEW AGAINST UK RGP

Claim 2.1.4.4: The PSA model is broadly aligned with UK RGP in terms of level of detail, accident sequence development, dependencies, and success criteria.

Argument 2.1.4.4-A1: The PSA has been developed using methods that are consistent with UK RGP, subject to further gap assessment and this will continue with the update to reflect the SMR-300 design.

Evidence for argument 2.1.4.4-A1:

Review of Current SMR-160 PSA against PSA Best Practice [44]

Provides a review of the SMR-160 Level 1 At-Power Internal Events PSA by Holtec International against UK RGP to inform the development of an acceptable PSA for the SMR-300 post GDA Step 2.

SMR-160 Level 1 At-Power PSA Gap Assessment [95]

The SMR-160 L1 At-Power PSA was compared against each Supporting Requirement set forth for the seven Technical Elements identified in Section 2-2 of the ASME/ANS RA-S-1.1–2022 [68].

SMR-160 Level 2 At-Power PSA Gap Assessment [96]

The L2 Internal Events At-Power Probabilistic was compared against each Supporting Requirement set forth for each of the six Technical Elements that comprise the L2 PSA, as identified in Section 2-1 of the ASME/ANS RA-S-1.2-2019 [59].

SMR-160 Low Power and Shutdown PSA Gap Assessment [97]

The SMR-160 LPSD PSA was compared against each Supporting Requirement set forth for the eight Technical Elements identified in part 3 of ANS/ASME-58.22-2014 [58] and each Supporting Requirement set forth for the six Technical Elements that comprise the as identified in Section 2-1 of the ASME/ANS RA-S-1.2-2019 [59].

SMR-160 Seismic PSA / Seismic Margins Analysis Gap Assessment [98]

The SMR-160 L1 and L2 PSA was compared against each Supporting Requirement set forth in Part 5 of the ASME/ANS RA-S-1.1–2022 [68]. The Internal Flood L2 PSA is fully based on L2 Internal Events At-Power PSA, therefore SMR-160 L 2 At-Power PSA Gap Assessment [96] is applicable.

SMR-160 Internal Fire PSA Gap Assessment [99]

The SMR-160 L1 and L2 PSA was compared against each Supporting Requirement set forth for the ten Technical Elements identified in Section 4-2 of the ASME/ANS RA-S-1.1–2022 [68]. The Internal Fire L2 PSA is fully based on L2 Internal Events At-Power PSA, therefore SMR-160 L 2 At-Power PSA Gap Assessment [96] is applicable.

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SMR-160 Internal Flood PSA Gap Assessment [100]

The SMR-160 L1 and L2 PSA was compared against each Supporting Requirement set forth for the three Technical Elements identified in Section 8-2 of the ASME/ANS RA-S-1.1–2022 [68]. The Internal Flood L2 PSA is fully based on L2 Internal Events At-Power PSA, therefore SMR-160 L 2 At-Power PSA Gap Assessment [96] is applicable.

SMR-160 External Flood PSA Gap Assessment [101]

The SMR-160 L1 and L2 PSA was compared against each Supporting Requirement set forth for the three Technical Elements identified in Section 8-2 of the ASME/ANS RA-S-1.1–2022 [68]. The External Flood L2 PSA is fully based on L2 Internal Events At-Power PSA, therefore SMR-160 L 2 At-Power PSA Gap Assessment [96] is applicable.

SMR-160 High Winds PSA Gap Assessment [102]

The SMR-160 L1 PSA was compared against each Supporting Requirement set forth for the three Technical Elements identified in Section 7-2 of the ASME/ANS RA-S-1.1–2022 [68]. The High Winds L2 PSA is fully based on L2 Internal Events At-Power PSA, therefore SMR-160 L 2 At-Power PSA Gap Assessment [96] is applicable.

16.7.1 ONR Minimum Probabilistic Limits

As mentioned in sub-chapter 16.4.2, a robust definition of UK PSA RGP has been established to inform the development of an acceptable PSA for the SMR-300 post GDA Step 2. This included the general minimum probabilistic limits set by the ONR that are not mandatory but strongly recommended. These limits are associated with the reliability of the software in a safety system and for credible reliability for a combination of related human actions detailed in NS-TAST-GD-046 [74] and NS-TAST-GD-063 [73] respectively.

While the SMR-160 Level 1 At-Power Internal Events PSA broadly complies with ONR's expectations for Human Performance Limiting Value (HPLV) where single or multiple Human Failure Events are claimed in the sequence; there is recognition in Part B Chapter 17 Human Factors [24] that Holtec's HRA process is not sufficiently mature to demonstrate full alignment with UK RGP (including justification of HPLVs). A Commitment has been raised within that chapter (see C_Huma_003). Furthermore, it should be noted that gaps between UK and US RGP have been identified, as discussed Part B Chapter 17 Human Factors Claim 2.1.7.1.

With regard to PSS, the failure probability determined for PSS software failure is below ONR's minimum probabilistic expectations and there are issues concerning the dependencies and screening of components from the CCF consideration. Therefore Claim 2.1.4.4 is not satisfied to a level commensurate for a Preliminary Safety Report. As part of the Commitment to update the PSA to reflect the SMR-300 post GDA Step 2 (see C_Faul_103, Part B Chapter 14 [5]) PSS software failure probability will be reviewed, in conjunction with a wider review and update of the PSS and DAS designs. The need to provide an adequately diverse PSS and DAS has been recognised and captured in Commitment C_C&I_082 (Part B Chapter 4 [19]). Target for Resolution of these Commitments is issue of UK Pre-Construction SSEC (see sub-chapter 16.9.3).

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16.7.2 Review Against Standards

Holtec International reviewed the SMR-160 PSA against the Supporting Requirements set forth in the ASME/ANS RA-S-1.1–2022 [68], ASME/ANS RA-S-1.2-2019 [59] and ANS/ASME-58.22-2014 [58]. The purpose of the Holtec International gap assessments was to identify Supporting Requirements that had not been addressed (partially or fully) in the SMR-160 PSA, to determine which Supporting Requirements should be addressed during the update to the SMR-300 PSA post GDA Step 2 and those that cannot be addressed until the site-specific stage. The comparisons showed that many of the Supporting Requirements were met at the Capability Category I (CC-I) level, which is the level of compliance required to support a Construction Permit Application (CPA). The Supporting Requirements not met were mainly as a result of lack of supporting design information that would only become available during the detailed design stages. Some other Supporting Requirements were determined not to be applicable because the plant is not built yet (i.e. there is not sufficient historical operating data available) and there is no ability to identify site-specific hazards or conduct walkdowns.

Holtec Britain conducted a separate review of the SMR-160 Level 1 At-Power Internal Events PSA against IAEA SSG-3 [67] and a new review against the ASME/ANS RA-S-1.1–2022 [68] to identify any additional differences required to meet UK PSA RGP. Overall, the review of the documentation on the SMR-160 Level 1 At-Power Internal Events PSA can be considered broadly aligned with international standards and with ONR expectations. In recognition concerns in relation to human error identification, screening analysis and the management of safety-critical systems of these concerns, a Commitment has been made (C_Huma_003) to present a strategy for the alignment of Holtec HRA process with UK RGP, post GDA Step 2 (see Part B Chapter 17 [24] for further details). Further reviews of the PSA are expected to be completed post GDA Step 2.

Holtec International is expected, as part of licensing, to complete a thorough peer review of the SMR-300 PSA. Following this, Holtec Britain will review the US peer assessment and have an external team conduct a targeted peer assessment on aspects that are different for the UK post-GDA Step 2. Therefore, a Commitment has been made (C_PSA_123) to undertake an IPR of the SMR-300 (see sub-chapter 16.4.1 and 16.9.3). Target for Resolution is Issue of UK Pre-Construction SSEC.

16.7.3 CAE Summary

Reviews of the SMR-160 PSA against the Supporting Requirements set forth in the ASME/ANS RA-S-1.1–2022 [68], ASME/ANS RA-S-1.2-2019 [59] and ANS/ASME-58.22-2014 [58], were met at the CC-I level, which is the level of compliance required to support a CPA. The Supporting Requirements not met were mainly as a result of lack of the input information that would only become available during the detailed design stages or when the plant has been built and operated for a sufficient length of time.

Overall, the review of the documentation on the SMR-160 L1 At-Power Internal Events PSA can be considered aligned with international standards and with most of ONR expectations. Some differences have been identified and will be considered in the wider update of the SMR-300 PSA.

Additionally, there are identified issues with the failure probability determined for PSS software claimed in the SMR-160 PSA being below ONR's minimum probabilistic expectations and related issues.

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Claim 2.1.4.4 is therefore not satisfied to a level commensurate for a PSR for Step 2 GDA. However, appropriate Commitments (C_Faul_103, C_C&I_082, C_PSA_123, C_Huma_003) have been made to address and close these gaps post GDA Step 2.

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16.8 PSA RESULTS AND INSIGHTS

Claim 2.1.4.5: PSA has been used to inform the design processes, including modifications, risk, and ALARP outcomes, as well as supporting the Design Basis Assessment (DBA) and Beyond Design Basis Assessment (BDBA).

Claim 2.1.4.5 has been further decomposed into five arguments. The PSA results are presented in terms of major contributors to mean CDF and LRF (A1) and confidence has been provided that the risk of the design to the public will be acceptable (A2). A PSA review of the PFS has been undertaken to support DBA and BDBA (A3). Insights from the SMR-160 PSA sensitivity studies and PSAs associated with legacy plants have been used to influence the SMR-300 design (A4). Design alternatives have been evaluated based on the risk reduction potential to inform the transition from the SMR-160 to the SMR-300 design (A5).

16.8.1 Major Contributors Identified for ALARP Assessment

[REDACTED]

16.8.1.1 Insights from the SMR-160 L1 'At Power' PSA

[REDACTED]

Table 4: SMR-160 Risk Significant L1 at Power Core Damage Sequences

[REDACTED]

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[REDACTED]

Figure 1: SMR-160 At Power Initiator Contribution to CDF

16.8.1.2 Insights from the SMR-160 L2 'At Power' PSA

[REDACTED]

Table 5: SMR-160 Risk Significant at Power Large Release Sequences

[REDACTED]

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[REDACTED]

Figure 2: SMR-160 At Power Initiator Contribution to LRF

16.8.1.3 Insights from the SMR-160 L1 LPSD PSA

[REDACTED]

Table 6: SMR-160 Risk Significant LPSD Core Damage Sequences

[REDACTED]

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[REDACTED]

Figure 3: SMR-160 LPSD Initiator Contribution to CDF

[REDACTED]

Figure 4: SMR-160 LPSD POS Contribution to CDF

16.8.1.4 Insights from SMR-160 L2 LPSD PSA

[REDACTED]

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Table 7: SMR-160 Risk Significant LPSD Large Release Sequences

[REDACTED]

[REDACTED]

Figure 5: SMR-160 LPSD POS Contribution to LRF

16.8.2 ONR Safety Assessment Principles (SAP) Risk Targets

[REDACTED]

16.8.2.1 Background

[REDACTED]

16.8.2.2 SAP Target 7 – Individual Risk of Fatality (Public)

[REDACTED]

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16.8.2.3 SAP Target 8 – Adult Dose Bands (Public)

[REDACTED]

16.8.2.4 SAP Target 9 – Contribution of 100 or More Fatalities

[REDACTED]

16.8.2.5 SAP Targets 5 and 6 (Site Worker Risks)

[REDACTED]

16.8.2.5.1 SAP Target 5 – Individual Risk of Fatality (Site Worker)

[REDACTED]

16.8.2.5.2 SAP Target 6 - Adult Dose Bands (Site Worker)

[REDACTED]

16.8.2.6 Other Considerations

[REDACTED]

16.8.2.7 Summary

[REDACTED]

16.8.3 Observations from Review of PFS

[REDACTED]

16.8.3.1 Evidence for Claim 2.1.4.5–A3:

[REDACTED]

16.8.4 Insights from PSA Sensitivity Studies

[REDACTED]

16.8.4.1 Evidence for Claim 2.1.4.5–A4:

[REDACTED]

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[REDACTED]

Figure 6: 'Sensitivity Studies PSA 'At Power' Initiating events Contribution to CDF

16.8.5 Identification of SMR-300 Design Alternatives

[REDACTED]

16.8.5.1 Evidence for Claim 2.1.4.5–A5:

[REDACTED]

16.8.6 CAE Summary

[REDACTED]

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16.9 CHAPTER SUMMARY AND CONTRIBUTION TO ALARP

This sub-chapter provides an overall summary and conclusion of the PSA chapter and how this chapter contributes to the overall demonstration of ALARP for the generic SMR-300. Part A Chapter 5 Summary of ALARP and SSEC [4] 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.
    - Options Considered to Reduce Risk.
- GDA Commitments.
- Conclusion.

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

16.9.1 Technical Summary

PSR Part B Chapter 16 aims to demonstrate the following Level 3 claim to a maturity appropriate for a PSR:

Claim 2.1.4: The Probabilistic Safety Assessment (PSA) demonstrates that the design of the generic Holtec SMR-300 is balanced such that risk is tolerable and As Low As Reasonably Practicable (ALARP).

At this stage the SMR-300 design is in its concept phase with few detailed drawings. The current PSA reflects the SMR-160 design and radionuclide inventories whereas the GDA application refers to the SMR-300 design which although identical in concept has some key differences. The focus of this chapter at PSR is around identifying and managing any gaps to ensure the validity of the PSA claims and arguments can be confidently met during the update of the PSA to reflect the SMR 300 design and UK good practice.

While the SMR 160 PSA is based on good practice accepted in the US and internationally there are recognised differences between the approach used in the UK and that used in the US. The differences identified [44] concerning, for example PSA scope (fuel route / fuel handling PSA), or need to a Level 3 PSA, are reflected in the Commitments made in this chapter (C_Faul_103, C_PSA_123). This provides confidence that adherence to US and international good practice will be maintained, as well as compliance with UK RGP, during SMR-300 PSA development post GDA Step 2 to support a UK site licence application.

Best-estimate methods, models, computer codes and data have been used to an extent consistent with the level of detail available. Uncertainty has been classified into aleatory uncertainty and epistemic uncertainty and the impact on the PSA assessed. Both the CAFTA software version 6.0b and MELCOR computer code version 2.2 were validated for L1 and L2 PSA modelling. There is a verification and validation plan for the RELAP5-3D computer code to assess the evaluation model post GDA Step 2.

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The current PSA directly relates to the existing SMR-160 design and will be updated to reflect the SMR-300 post GDA Step 2 (see Commitment C_Faul_103). Sensitivity studies undertaken to capture expected design changes for SMR-300 and to address known differences in UK regulatory expectation provides confidence that the CDF for the SMR-300 is likely to be of the same order of magnitude as the CDF for the SMR-160, recognising the Commitment to address C&I architecture issues to ensure appropriate diversity is provided (C_C&I_082).

Potential initiating events were identified based on generic industry and plant-specific system and design features. The PIEs and IEFs in the PFS [89] were reviewed and assessed for inclusion in the UK PSA scope post GDA Step 2. Holtec Britain has provided a plan on hazard screening based on UK RGP and additional hazards were identified that will need to be included or revisited in the PSA for the SMR-300 post GDA Step 2.

It is noted, that, to date, there has not been a systematic ‘bottom up’ hazard identification activity undertaken for the SMR-300 PSA which would align with UK RGP. Furthermore, it is recognised that the bounding of IEs in the PSA does not currently align with accepted UK RGP. As discussed in Sub-Section 16.6.1, a Commitment is raised, C_Faul_103 to undertake, as necessary, supplemental safety activities to incorporate the full scope UK SMR-300 design to align with UK RGP and regulatory expectation.

Severe accident phenomena which have a potential to threaten the RCS pressure boundary and/or containment structural integrity have identified from regulatory and industry guidance. The L2 PSA is bounding in that it does not credit mitigating systems or capabilities that are relevant only to a radionuclide release. The Internal Events PSA addresses risks associated with full power operation and other POSs for both L1 CDF and L2 LRF.

For Revision 1 it is deemed that the maturity of the safety justification presented in this chapter (Part B Chapter 16) is appropriate for a PSR. The SMR-160 PSA provides confidence that Claim 2.1.4, will be substantiated for the SMR-300 in future safety submissions and that risks will be evaluated as tolerable and ALARP.

16.9.2 ALARP Summary

The information provided illustrates that appropriate use of PSA has been, and will be, made in supporting the demonstration that risks are tolerable and ALARP. The SMR-160 PSA has provided a number of ways of gaining insight into nuclear risks with a view to providing confidence that the overall nuclear risk will be ALARP:

- Provided confidence that risks to the public will ultimately be acceptable by comparison with the numerical targets described in the ONR SAPs.
- Showed the balance of risk contributors so that the highest risk contributors can be addressed first, maximizing the benefit of any proposed design changes.
- Allows the risk reduction potential of different options to be able to be compared, thus giving insight into which option, or combination of options, is likely to give the biggest risk improvement.
- Allows the risk reduction potential of a proposed design change to be assessed as part of the assessment of gross disproportion, that is, it allows the risk reduction to be compared with the cost of the design change in terms of effort, time or money.

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16.9.2.1 Demonstration of RGP

Sub-chapter 16.4 has shown that the generic SMR-160 methodology for PSA has been undertaken based on the identified relevant US and international standards, regulatory requirements, standards, and industry best practices. Contradictions between requirements have also been considered and those deemed to be the most restrictive are prioritised in the development of the PSA. Also, differences were identified between the US and UK regulatory approaches to identify the additional considerations that are necessary for meeting UK PSA RGP set out in relevant ONR PSA TAGs and IAEA guidance.

Sub-chapter 16.7 has shown that the reviews of the SMR-160 PSA against the Supporting Requirements set forth in the ASME/ANS RA-S-1.1–2022 [68], ASME/ANS RA-S-1.2-2019 [59] and ANS/ASME-58.22-2014 [58], were met at the CC-I level, which is the level of compliance required to support a CPA. The Supporting Requirements not met were mainly as a result of lack of the input information that would only become available during the detailed design stages or when the plant has been built and operated for a sufficient length of time.

Whilst the SMR-160 Level 1 At-Power Internal Events PSA can be considered broadly aligned with international standards and with ONR expectations; there are differences identified that will need to be addressed in the update of the SMR-300 UK PSA, post GDA Step 2.

16.9.2.2 Evaluation of Risk and Demonstration Against Risk Targets

Sub-chapter 16.8.1 provides the results and insights from the SMR-160 L1 and L2 PSA. The discussion provided has demonstrated a robust process for understanding the risk outcomes and major risk contributors using FV and RAW importance measures. Therefore, addressing contributors in order of decreasing FV is most likely to lead to a design whose risks are ALARP, provided the costs of improving the contributor are not grossly disproportionate. Similarly, contributors with the highest RAW have the largest contribution to overall risk and addressing those with the highest RAW first identifies contributors with the greatest potential to achieving a design whose risks are ALARP. This process is being taken forward during the transition to the SMR-300 design.

Sub-chapter 16.5.4 identifies specific differences between the SMR-160 and SMR-300 design that impact the PSA. The change in SMR-160 to SMR-300 power levels results in a corresponding change in the specification of component duties and sizes, system flow rates, volumes, and water inventories etc, however, the overall design concept (types of system, number of trains, components, and associated success criteria for accident mitigation) generally remains unchanged. Likewise, the internal events accident sequence development and overall risk profile are expected to be similar.

To support this conclusion, specific differences between the SMR-160 and SMR-300 that impact the PSA have been identified and included in a set of SMR-160 sensitivity studies performed during GDA Step 2. Sub-chapter 16.8.4 presents the results of the sensitivity studies and provide confidence that the CDF for the SMR-300 is likely to be of the same order of magnitude as the CDF (increase of 22%) for the SMR-160 once the expected design changes have been incorporated into the PSA model. However, due to 100% of VLOCA sequences currently assumed to result in containment failure and LR, the corresponding total LRF for At Power internal events is expected to increase by ~190%.

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Sub-chapter 16.8.2 uses the results from the SMR-160 PSAs and the PSA Sensitivity Studies to provide confidence that the risk of the SMR-300 design to the public will be likely acceptable by comparison with Numerical Targets 7, 8 and 9 as defined in the ONR SAPs.

16.9.2.3 Options Considered to Reduce Risk

Sub-chapter 16.8.5 documents the reliability assessments undertaken in support of system design development for the SMR-300. These assessments were undertaken to provide insights to system designers that may result in system design changes to improve the reliability of the system or provide assurance that the system level design meets reliability requirements. System level reliability analysis and importance calculations are provided for the SSCs modelled in the following system level fault trees:

- Secondary Decay Heat Removal System [103].
- Automatic Depressurization System [104].
- Instrument and Service Air System [105].
- Reactor Trip and Engineered Safety Features Functions [106].
- Reactor Coolant System [107].
- Diverse Actuation System [108].
- Service Water System [109].
- Passive Core Makeup Water System [110].
- AC Power System [111].
- Spent Fuel Pool Cooling System [112].
- DC Power System [113].
- Main Steam System [114].
- Primary Decay Heat Removal System [115].
- Circulating Water System [116].

A number of Design Challenges have been raised during the course of GDA Step 2, as set out in Part A Chapter 2 [6]. These have the potential to further develop and evolve the SMR-300 design and PSA will be appropriately used to support Design Challenge resolution and inform their outcomes. Development of future PSA models beyond GDA Step 2 will also reflect subsequent UK reference designs which incorporate Design Challenge outcomes (see C_Faul_103).

These challenges follow the process for the assessment of the risk reduction options presented in GDA Design Management Process [117].These challenges follow the process for the assessment of the risk reduction options presented in GDA Design Management Process [117].

In addition, Commitments made in this chapter and discussed in the subsequent sub-chapter will be reflected in the SMR-300 UK PSA, post GDA Step 2, and support the ongoing identification of risk significant aspects of the design and operation of the SMR-300.

16.9.3 GDA Commitments

Holtec recognise that differences exist between US and UK requirements, with regards to the scope, methodology and extensiveness of the safety assessment applied to the SMR-300 design. The Preliminary Safety Analysis Report (PSAR) for the Palisades SMR-300 design will reference safety analyses that will be repurposed for any future UK SSEC.

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Sub-chapter 16.1.1 notes, as per ONR-GDA-GD-007 [3], a full scope PSA is required for Step 2 recognising that the depth and level of detail will not be fully captured given the stage of design. Therefore, a formal design challenge was raised by Holtec Britain on the completeness of the PSA [76], to decide on whether it would be more beneficial to have a separate UK PSA model to address differences in US and UK regulatory requirements (e.g. the U.S.NRC not requiring PSA for fuel route or SFP). The Design Adaptation Committee followed GDA Design Management Process [117] to review the design challenge and made the decision to continue with a single SMR-300 PSA model and proceed with a separate UK variant closer to the site-specific stage when the UK adapted generic design becomes available. The following Commitments, raised in this, and other PSR chapters reflect the above discussion:

C_PSA_123: PSA methodologies employed on the SMR-300 require further development to fully align with UK regulatory expectations, including the requirement for Independent Peer Review of the developed PSA models. A Commitment is raised to perform Independent Peer Review of the SMR-300 PSA to support the issue of the UK Pre-Construction SSEC. Target for Resolution – Issue of UK Pre-Construction SSEC.

The following GDA Commitment identified in Part B Chapter 14 [5] captures key requirements for development of the PSA beyond Step 2 GDA, as further discussed in sub-chapters 16.1.1, 16.4.2, 16.5.4, 16.6.1, 16.6.2, 16.6.3, 16.6.4, 16.7.1, 16.8.2.5, 16.8.3.1, 16.8.5.1 and 16.9.1:

C_Faul_103: Holtec commit to ensuring that the repurposing of the US safety analyses undertaken for the Palisades SMR-300 design also considers and undertakes, as necessary, supplemental safety assessment to appropriately address UK expectations and good practice. This supplemental assessment should incorporate the full scope UK SMR-300 design and will be targeted to ensure a holistic and comprehensive approach across the recognised safety assessment disciplines. Future UK SSEC is therefore expected, as a minimum, to encompass:

- Completion of the identification of PIEs, within the full scope UK SMR-300 design.
- Harmonization between this initiating event list for use in both deterministic and probabilistic assessments.
- Extension of the scope of PSA to assess the SMR-300 design and operation to Level 3 PSA; this will include all sources of radionuclide release and operations (such as the SFP) and all potential initiating events (e.g. Internal Hazards, External hazards).
- Development of a UK-aligned set of design basis faults.
- An updated UK Fault and Protection Schedule, which covers all design basis faults for the SMR-300.
- UK DBAA studies to:
    - Identify UK aligned expectations for safety function categorisation and SSC classification for each bounding fault.
    - Demonstrate, supported by appropriately verified and validated UK DBAA, that the design can safely mitigate all design basis faults.
    - Undertake supporting radiological consequence analysis to demonstrate the residual risks are tolerable and ALARP.
- UK-aligned Severe Accident studies, informed by the PSA and DBAA, to ensure that the facility can be brought into a long term safe, stable state.
- Incorporate Human Factor Engineering analysis (including Human Reliability Analysis) throughout DBAA/PSA/SAA.

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Finally, the following Commitments raised elsewhere in the PSR also require consideration in the development of the PSA post Step 2 GDA:

C_Huma_003 (see sub-chapter 16.4.1, 16.7.1, 16.7.2): The SMR-300 design, and underlying design processes demonstrate that human risks are systematically identified, proportionately assessed, and that the risk contribution from operator actions is tolerable. It is noted that US Human Reliability Assessment (HRA) outputs for the SMR-300 will not be available for review within GDA timeframes and as such cannot be integrated with UK safety assessment processes which HRA must effectively support and integrate with. A Commitment is raised to develop a HRA Strategy for the UK SMR-300 design, describing how to make use of the processes for deterministic and probabilistic safety assessment used by the U.S.NRC, as well as information relating to the Treatment of Important Human Actions in UK safety assessment. Target for Resolution - Issue of UK Pre-Construction Safety Report.

C_C&I_082 (see sub-chapters 16.7.1, 16.8.4.1): Design Challenge Paper 'I&C Architecture' (HI-2240612) associated with PSR Part B Chapter 4 Claim 2.2.6.2 'The I&C system design incorporates Defence in Depth to protect against anticipated operational occurrences and accident conditions' is with the Design Authority for Design Decision. This Design Challenge relates to differences between the US and UK Regulatory regimes in the I&C discipline and the potential need for a design change to the DAS.

A Commitment is raised to progress this Design Challenge through the Design Management process (HPP-3295-0017-R1.0) to completion. Notably, the DAS design will be modified to use a non-computerised, simple hardware-based DAS which is adequately diverse from the technology used in the PSS. The use of shared equipment between the PSS and DAS will be reviewed and design options considered. The design of the output interface modules (PIM) will also be reviewed and justified. Target for Resolution - Issue of UK Pre-Construction SSEC.

16.9.4 Conclusion

Part B Chapter 16 presents the approach undertaken for PSA in support of GDA Step 2, and a summary of the results and important contributors to risk for in reactor faults. It identifies the relevant claims, arguments and currently available evidence that form the basis of the safety case for the PSA topic to a maturity appropriate for a PSR.

The chapter acknowledges that the SMR -300 PSA is currently under development by Holtec International and that this PSA will be available, post Step 2 GDA, to support future licensing activities. It is also accepted that a UK variant of the generic SMR-300 PSA will ultimately be produced. As such the chapter primarily draws on the SMR-160 L1 and L2 PSAs, and further gap assessment and sensitivity analysis to evidence the claims.

It is concluded that the chapter claims have been broadly met with the exception of the following:

- Claim 2.1.4.3 (sub-chapter 16.6), in particular regarding the scope of the PSA which at this stage, given the maturity of the design, does not address SFP and fuel route hazards, or Internal and External hazards; and the demonstration of completeness of the IE hazard identification process.
- Claim 2.1.4.4 (sub-chapter 16.7), in particular regarding application of UK RGP, regarding the categorisation and assessment of components at risk from CCF.

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However, associated Design Challenges and GDA Commitments have been raised to close all gaps identified within the chapter and as the design and safety case continue to be developed, further evidence will be provided to substantiate these claims.

The following key points are also noted:

- The SMR 160 PSA is based on good practice accepted in the US and internationally, however there are recognised differences between the approach and methodologies used in the UK and that used in the US. These differences have been identified and a robust definition of UK RGP has been defined [44].
- There is a complete L1 and L2 PSA for the SMR-160 design and these assessments cover internal reactor faults for At Power and LPSD modes of operation. The key design differences between SMR-300 have been identified and have been assessed in a set of PSA sensitivity studies [37], which also address any changes required to align with UK RGP. The results have provided confidence that the risk profile for the SMR-300 is broadly aligned with the SMR-160, noting the Commitments to provide a SMR-300 PSA for UK licensing and to address C&I architecture and diversity in design (see sub-chapter 16.9.3).
- The SMR-160 PSAs have been used to identify the important contributors to risk, with further confirmatory evidence provided in the Sensitivity Studies [37]. System reliability analysis has also been undertaken in support of design development.
- A high level comparison of the PSA results (based on the SMR-160 analysis supported by the sensitivity studies) against the ONR numerical targets has provided confidence that these can be met for the twin-unit SMR-300 design.

The ALARP considerations are discussed in the context of the overall SMR-300 design in an overarching ALARP summary statement in Part A Chapter 5 Summary of ALARP and SSEC [4].

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16.11 LIST OF APPENDICES

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Appendix A PSR Part B Chapter 16 CAE Route Map

Table 8: PSR Part B Chapter 16 CAE Route Map

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