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UK HPR1000 GDA Project

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Preliminary Safety Report

Chapter 14

Probabilistic Safety Assessment

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14.1 List of Abbreviations and Acronyms

AAD	Startup and Shutdown Feedwater System [SSFS]
ALARP	As Low As Reasonably Practicable
ASEP	Accident Sequence Evaluation Program
ATWS	Anticipated Transient Without Scram
BOP	Balance of Plant
BSL	Basic Safety Level
BSO	Basic Safety Objective
CDF	Core Damage Frequency
CET	Containment Event Tree
DCH	Direct Containment Heating
EHR	Containment Heat Removal System [CHRS]
FCG	Fangchenggang Nuclear Power Plant
FDF	Fuel Damage Frequency
GCT	Turbine Bypass System [TBS]
GDA	Generic Design Assessment
GIS	Graphical Information System
HPR1000	Hua-long Pressurized Reactor
HPR1000 (FCG3)	Hua-long Pressurized Reactor under construction at Fangchenggang Nuclear Power Plant Unit 3
HRA	Human Reliability Analysis
IAEA	International Atomic Energy Agency
IE	Initiating Event
LPSD	Low Power and Shutdown
LRF	Large Release Frequency
MCCI	Molten Core Concrete Interaction
MSQA	Management System and Quality Assurance
ONR	Office for Nuclear Regulation

PDS	Plant Damage State
PHE	Public Health England
POS	Plant Operating State
PSA	Probabilistic Safety Assessment
PWR	Pressurized Water Reactor
RHR	Residual Heat Removal System
RRI	Component Cooling Water System [CCWS]
/ry	per reactor year
SAPs	Safety Assessment Principles
SEC	Essential Service Water System [ESWS]
SF	Safety Function
SG	Steam Generator
SFIS	Spent Fuel Interim Storage
SFP	Spent Fuel Pool
SPAR-H	Standardized Plant Risk Analysis-Hum an Reliability Analysis Procedure
SPSA	Seismic PSA
SSCs	Structures, Systems and Components
UK HPR1000	The UK Version of the Hua-long Pressurized Reactor

System codes (XXX) and system abbreviations (YYY) are provided for completeness in the format (XXX [YYY]), e.g. Component Cooling Water System (RRI [CCWS]).

14.2 Introduction

This chapter presents information to support understanding that the UK Version of the Hua-long Pressurized Reactor (UK HPR1000) design will be developed in an evolutionary manner, using robust design processes, building on relevant good international practice, to achieve a strong safety and environmental performance. This chapter will demonstrate that the Probabilistic Safety Assessment (PSA) is used to understand the nuclear safety risk profile, and to inform decisions on improvements to the design. Through the course of the licensing process the PSA will be developed to support the safety analysis that underpins the safety performance of the UK HPR1000.

The PSA that will be developed for the UK HPR1000 supports the following high level objective:

“The design and intended construction and operation of the UK HPR1000 will protect the workers and the public by providing multiple levels of defense to fulfill the fundamental safety functions”.

This chapter will demonstrate the following:

The PSA is used to understand the nuclear safety risk profile, and to inform decisions on improvements to the design.

The design version of the UK HPR1000 for the Generic Design Assessment (GDA) has not been declared yet. The design will be based on the version of HPR1000 (FCG3).

During the course of GDA, the PSA for UK HPR1000 will be developed from the Level 1 and Level 2 PSA used in support of the HPR1000 (FCG3) safety analysis into a full scope PSA (Levels 1, 2 and 3), in line with the design evolution of the UK HPR1000, to consider all operating modes, and the full range of fault conditions. Since the PSA for the HPR1000 (FCG3) will be the basis for the eventual PSA for the UK HPR1000, that this basis is sound will need to be demonstrated.

The strategy adopted for this chapter is to take full cognizance of the work undertaken for HPR1000 (FCG3), the approach adopted therein and the insights arising therefrom and to use this to demonstrate the intent for UK HPR1000.

In support of the above, the remainder of this chapter is structured to consider:

a) Sub-chapter 14.3 discusses the methodologies used in the development of the HPR1000 (FCG3) PSA ;

b) Sub-chapter 14.4 presents the key findings arising from the HPR1000 (FCG3) PSA;

c) Sub-chapter 14.5 outlines the requirements to support development of the UK HPR1000;

d) Sub-chapter 14.6 discusses the evidence presented to support the claims outlined above.

The PSA is a key chapter in the presentation of the analysis of nuclear safety. The PSA models reflect, and provide support to, the design of plant and systems for which associated chapters are presented in the PSR. Additionally, the PSA utilizes information from the activities and processes covered by the chapters summarized in T-14.2-1.

T-14.2-1 Key interfaces between the PSA and other areas of the PSR

Chapter	Title	Interfaces with the PSA
1	Introduction	Presentation of the overall objectives for the PSR.
3	Generic Site Characteristics	Presentation of generic site data for hazards (e.g. flood, seismic).
4	General Safety and Design Principles	PSA support to validation of Safety Measure reliability (with Core Damage Frequency (CDF) and Large Release Frequency (LRF) targets).
12	Design Basis Conditions Analysis	Presents the Fault Schedule, which can support Level 1 and Level 2 PSA.
13	Design Extension Conditions & Severe Accident Analysis	Presents the beyond design basis analysis, which is an input to Level 2 PSA, covering contributions to containment breach frequency.
15	Human Factors	Provision of/support to Human Error Probability assessments.
18	External Hazards	Definition and characterization of the hazards, to support the hazards elements of the PSA.
19	Internal Hazards	Definition and characterization of the hazards, to support the hazards elements of the PSA.
20	Management System and Quality Assurance (MSQA) & Safety Case Management	Outlines the safety case process, and includes the role of the PSA in supporting the analysis of risk levels in support of the As Low As Reasonably Practicable (ALARP) process.
21	Reactor Chemistry	Source term information from chapter 21 will support Level 2 PSA.
22	Radiological Protection	The result of Level 3 PSA will be performed to compare with the Targets 7, 8 and 9 of the

Chapter	Title	Interfaces with the PSA
		Safety Assessment Principles (SAPs) which are related to the radiation protection.
26	Environment	General site information including population and metrological data that supports the environmental impact assessment could be used to support the PSA Level 3 analysis.

14.3 HPR1000 (FCG3) PSA Methodology

This sub-chapter outlines the scope and methods adopted for the various elements of the HPR1000 (FCG3) PSA models.

The HPR1000 (FCG3) PSA has been developed in accordance with internationally accepted practices, a comprehensive set of references were used to shape the analysis (T-14.3-1 refers). The principal references used were:

Chinese standards:

a) NB/T 20037.1-2011 - Probabilistic Safety Assessment for Nuclear Power Plant Applications Part 1: Internal Events Level 1 PSA for Power Operation;

b) NB/T 20037.2-2012 - Probabilistic Safety Assessment for Nuclear Power Plant Applications Part 2: Internal Events Level 1 PSA for Low Power and Shutdown states;

c) NB/T 20037.3-2012 - Probabilistic Safety Assessment for Nuclear Power Plant Applications Part 3: Internal Flooding;

d) NB/T 20037.4-2013 - Probabilistic Safety Assessment for Nuclear Power Plant Applications Part 4: Internal Fire;

e) NB/T 20037.5-2013 - Probabilistic Safety Assessment for Nuclear Power Plant Applications Part 5: Seismic.

The above standards are all derived from the US and international standards referenced below.

International technical standards and guidelines:

a) International Atomic Energy Agency (IAEA), Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants Specific Safety Guide. IAEA Safety Standards SSG-3. 2010;

b) IAEA, Development and Application of Level 2 PSA for Nuclear Power Plant. IAEA Safety Standards SSG-4. 2010;

c) NUREG/CR-6850, EPRI/NRC-RES Fire PRA methodology for nuclear power facilities, Final Report, NUREG/CR-6850, EPRI 1011989.

T-14.3-1 Standards and Guidelines supporting the development of the HPR1000 (FCG3) PSA

Standards and Guidelines	
1.	IAEA. Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants Specific Safety Guide. IAEA Safety Standards SSG-3. 2010.
2.	IAEA. Development and Application of Level 2 PSA for Nuclear Power Plant. IAEA Safety Standards SSG-4. 2010.
3.	The National Energy Administration. Probabilistic Safety Assessment for Nuclear Power Plant Applications - Part 1: Level 1 PSA for Internal Events at-power. NB/T20037.1-2011. 2011.
4.	The National Energy Administration. Probabilistic Safety Assessment for Nuclear Power Plant Applications - Part 2: Level 1 PSA for Internal Events at-low Power and Shutdown. NB/T20037.2-2012. 2012.
5.	The National Energy Administration. Probabilistic Safety Assessment for Nuclear Power Plant Applications - Part 3: Flooding, NB/T 20037.3. 2012.1.
6.	The National Energy Administration. Probabilistic Safety Assessment for Nuclear Power Plant Applications - Part 4: Fires (draft for approval), NB/T 20037.4. 2012.
7.	The National Energy Administration. Probabilistic Safety Assessment for Nuclear Power Plant Applications Part 5: Seismic, NB/T 20037.5. 2013.
8.	EPRI. Methodology for Assessment of Nuclear Power Plant Seismic Margin. EPRI NP-6041-SL. 1991.
9.	EPRI. Seismic Probabilistic Risk Assessment Implementation Guide. EPRI TR 3002000709, 2013.
10.	EPRI/NRC-RES. Fire PRA Methodology for Nuclear Power Facilities, NUREG/CR-6850, EPRI 1011989, 2005.
11.	Office of Nuclear Regulatory Research. A Framework for Low Power/Shutdown Fire PRA, NUREG/CR-7114, 2011.
12.	Office of Nuclear Regulatory Research. The SPAR-H Human Reliability Analysis Method, NUREG/CR-6883, 2004.
13.	Sandia National Laboratories. Accident Sequence Evaluation Program Human Reliability Analysis Procedure, NUREG/CR-4772, 1987.
14.	EPRI. Fire Probabilistic Risk Assessment Methods Enhancements, EPRI 1019259, 2009.

The software used in the production of the HPR1000 (FCG3) PSA (Level 1 and Level 2) is RiskSpectrum PSA program (Version 1.3.2) of Scandpower AB.

At this stage, a Level 3 PSA of UK HPR1000 is still to be developed and consideration of possible options for this is outlined in sub-chapter 14.5.6.

14.3.1 Level 1 PSA

The HPR1000 (FCG3) Level 1 PSA evaluates core damage risk of the reactor and the plant operation states include Power operation and Low Power and Shutdown (LPSD) states. The scope of the HPR1000 (FCG3) Level 1 PSA encompasses:

a) Plant Faults (internal events);

b) Hazards (external events):

1) Internal Hazards (internal fire and internal flooding);

2) External Hazards (man-made and natural).

c) The following Fuel Route (Spent Fuel Pool) operations:

1) Short term storage of assemblies in the SFP;

2) Transfer of irradiated fuel between the SFP and the reactor, and vice-versa.

The approach adopted for the HPR1000 (FCG3) PSA is outlined in the remainder of this sub-chapter.

14.3.1.1 Plant Faults (Internal Events) Level 1 PSA

The scope of Plant Faults contains Power Operation and LPSD. A key difference between the approach adopted for the production of LPSD PSA and that adopted for Power Operation is the need to recognize a number of different plant operating states (POS). The definition of the different POSs reflect different power plant configurations whose operational parameters are relatively stable and the nature of the risk impact is different from those of other configurations. A total of seven POSs (the defined POSs are only used for PSA analysis) have been used in the HPR1000 (FCG3) PSA and these are presented in T-14.3-2.

T-14.3-2 Definition of Plant Operating States

POS	Name
POSA	Power operation
POSB	Shutdown condition of Steam Generator (SG) cooling mode
POSC	Shutdown condition of Residual Heat Removal System (RHR) cooling mode
POSD	Maintenance cold shutdown when manholes are closed
POSE	Maintenance cold shutdown when manholes are open
POSF	Refueling cold shutdown mode and Reactor completely discharged mode
POSG	MID-LOOP[1] condition

[1] Mid-LOOP is a normal operational state for purification and degasifier service during maintenance cold shutdown.

14.3.1.1.1 Power Operation

The internal events PSA undertaken for HPR1000 (FCG3) has been developed in accordance with the methods outlined in the relevant standards and for which the key steps can be summarized as follows:

a) Collection of plant information;

b) Identification of Initiating Events (IEs) and estimation of their frequencies;

c) Analysis of accident sequences;

d) Establishment of success criteria;

e) System reliability analysis;

f) Human reliability analysis;

g) Data analysis;

h) Quantification of accident sequences;

i) Uncertainty and sensitivity analysis;

j) Documentation.

The following provides a brief outline of the main process adopted in the production of the HPR1000 (FCG3) PSA and provides concluding remarks on the methods adopted.

14.3.1.1.2 IE Identification

The main references for the selection and grouping of IEs for internal events of FCG Unit 3 are as follows:

a) List of operating conditions for FCG Unit3;

b) List of common IEs for Pressurized Water Reactor (PWR) nuclear power plants (NUREG/CR-5750, Reference [1], NUREG/CR-6928, Reference [2]);

c) Related experiences from operations of CPR1000 series units, including Daya Bay Nuclear Power Plant and Ling'ao Nuclear Power Plant;

d) List of IEs in CPR1000 and Taishan Phase I.

T-14.3-3 IEs – Plant Faults

IE	
Code	**Description**
LLOCA	Large break (>180 cm^2) in coolant pipelines
MLOCA	Medium break (45-180 cm^2) in coolant pipelines
SLOCA	Small break (2-45 cm^2) in coolant pipelines

IE	
Code	**Description**
VR	Pressure vessel rupture
VLOC	Interface LOCA
LOCC	Loss of any train of Component Cooling Water System (RRI) [CCWS]
LOMFW	Loss of main feedwater
LOOP	Loss of offsite power
FLB	Break in main feedwater pipe
SLB	Break in main steam pipe
SGTR	Steam generator tube broken
LODC	Loss of direct current
ST	Secondary transient
PT	Other transients (primary)
DIL	Boron dilution
LORHR	Loss of RHR
LOAAD	Loss of AAD
CP	Cold overpressure

14.3.1.1.3 IE Quantification

Each IE frequency was derived from one of the following methods:

a) Reference to the common IE frequency data for PWR nuclear power plants (NUREG/CR-5750, Reference [1], NUREG/CR-6928, Reference [2], NUREG-1829, Reference [3]);

b) By developing system fault trees. Mainly for those IEs that are closely related to the system design (loss of Residual Heat Removal (RHR), the loss of the Component Cooling Water System (RRI [CCWS]) and the loss of Essential Service Water System (SEC [ESWS]), etc.);

c) Data from other plants (Daya Bay Nuclear Power Plant and Taishan Phase 1).

Table T-14.3-3 presents a list of IEs (plant faults) considered in the HPR1000 (FCG3) PSA.

14.3.1.1.4 Accident Sequence Analysis

The HPR1000 (FCG3) PSA adopts the small event tree/large fault tree approach. With this approach the inter-dependencies between front-line systems and support systems is addressed within the detail of the supporting fault trees and the number of Functional Events (nodes) on the event trees is much reduced. That is, only specific SFs are included in the functional event, such as front-line systems or human operations.

Event sequence, event structures and end states are defined for each IE category based on the expected response of mitigation systems. Success criteria have been established to determine the minimum set of trains or components that will successfully perform an intended function. For several success criterion and time availability for human error events, specific thermal and hydraulic calculations have been conducted.

14.3.1.1.5 Success Criterion

The ability of fundamental SFs to prevent core damage relies on the definition of minimum levels of performance of the supporting safety systems.

The success criterion for each safety system has been defined as the minimum level of performance required to achieve the SF, taking into account the specific features of each sequence.

Success criteria for Level 1 PSA have been defined for systems that fulfil the following functions:

a) Reactivity control;

b) Core cooling and;

c) Long-term heat removal.

14.3.1.1.6 System Reliability Analysis

Fault Tree Analysis has been used to model all system functions used to mitigate IEs.

The main failure modes considered include:

a) Equipment fault due to hardware fault;

b) System driving fault due to I&C fault (loss of, or spurious signal) or spurious operation by operator;

c) Power supply fault;

d) Pre-accident human error;

e) Test and maintenance errors that result in the unavailability of the system.

Supporting systems are modeled through the use of a transfer/ transmission gate.

14.3.1.1.7 Human Reliability Analysis

The Human Reliability Analysis (HRA) considered for HPR1000 (FCG3) PSA includes both pre-accident and post-accident HRA.

Pre-accident HRA deals with the latent errors performed during maintenance or testing of components. These errors, if left undiscovered, result in the unavailability of some components which could be necessary to mitigate an accident. Pre-accident HRA utilized the Accident Sequence Evaluation Program (ASEP), Reference [4] as the basis for evaluating the actions.

Post-accident HRA deals with the failure of operators to implement the actions necessary to mitigate an accident. Post-accident HRA utilized the Standardized Plant Analysis Risk-Human Reliability Analysis (SPAR-H), Reference [5] as the basis for evaluating the actions.

14.3.1.1.8 Data Analysis

HPR1000 (FCG3) is a newly designed PWR unit and, consequently, there is no specific operating experience data available.

The equipment reliability data, common cause parameters and IE frequency used in the current analysis are mainly based on the operating experiences of nuclear power plants. This data was selected on the basis that the data from United States of America reflects a rich source of operating experiences and the most complete statistic systems. According to item 656 in SAP, Reference [6] generic data can also be used in the PSA work.

The general data sources used in the HPR1000 (FCG3) PSA are as follows:

a) IE frequency: NUREG/CR-6928, Reference [2], NUREG/CR-5750, Reference [1], NUREG-1829, Reference [3], Operating experience from CGN;

b) Equipment reliability data: NUREG/CR-6928, Reference [2];

c) Equipment common cause parameters: NUREG/CR-5497 (2012), Reference [7].

14.3.1.1.9 Quantification of Accident Sequences

Accident sequence quantification has been undertaken using the RiskSpectrum PSA program (Version1.3.2) of Scandpower AB. This software is a PSA tool widely accepted and used in the nuclear industry throughout the world.

The probability truncation method is used for the quantitative calculations in the current analysis and the cut-off value is set at 1E-13. Parameter sampling is adopted for the uncertainty analysis and the number of samplings is 30,000.

14.3.1.1.10 Uncertainty and Sensitivity Analysis

Uncertainty analysis has been undertaken on all segment of PSA.

Specific examples include sensitivity studies undertaken to assess the benefits (potential risk reduction) arising from claiming Turbine Bypass System (GCT) [TBS] used for fast cool down at medium pressure and also the analysis of fewer than three (the baseline assumption) bleed and feed trains.

14.3.1.1.11 Documentation

The HPR1000 (FCG3) PSA is fully documented and much of the information presented in this chapter is a summary of that information.

14.3.1.1.12 Low Power and Shutdown Operation

The development of the Level 1 LPSD PSA for internal events adopts internationally accepted methods and complies with the requirements on PSA work for nuclear power plants at the design phase as specified in NB/T 20037.2-2012, Reference [8].

The methods used for event sequence analysis, system analysis, human reliability analysis, data analysis and technical elements such as software platform in the development of Level 1 PSA for internal events in LPSD conditions are the same as those used in power condition.

The methods used for selection and grouping of IEs in shutdown condition for FCG Unit 3 are the same as those used in power condition. The selection of the IEs modeled in each POS is dependent on the relevance of the at-power IEs to the plant conditions associated with the POS.

14.3.1.1.13 Sub-Chapter Conclusions

The Level 1 PSA for plant faults has been developed taking due cognizance of recognized standards and provides a strong foundation on which the UK HPR1000 PSA can be developed. The key findings arising from the development of the HPR1000 (FCG3) and areas for consideration for enhancement to support the development of the UK HPR1000 are identified in sub-chapters 14.4 and 14.5 respectively.

14.3.1.2 Internal and External Hazards Level 1 PSA

The method adopted for the hazards PSA is based on the guidelines presented in IAEA SSG-3, Reference [9].

Analysis of internal and external hazards can be considered to comprise three main steps:

a) Hazard identification to obtain a comprehensive list of external events;

b) Hazard screening to ensure that the subsequent assessment of these hazards is focused on those hazards that are considered relevant. This process screens out those hazards that are not applicable or not important;

c) Detailed analysis to characterize the hazards and to reflect the potential threat from these in the PSA models for important/significant hazards or evaluate their risks using other methods.

In addition to consideration of individual hazards, consideration has also been given to co-incident occurrence of hazards.

For considering hazards combination, the possible combination of hazards should be identified on the basis of the list of individual internal and external hazards.

The general approach used for the identification of a realistic set of combinations of hazards should be based on a systematic check of the dependencies between all internal and external hazards.

14.3.1.2.1 Hazard Identification

The hazard identification process adopted for HPR1000 (FCG3) PSA is derived from the list of IAEA SSG-3, Reference [9] and amended to reflect CGN's experience. It should be noticed that the list of hazards, especially external hazards, may be bound to be amended according to site specific data. The hazards list is presented in T-14.3-4 and T-14.3-5.

T-14.3-4 List of Internal Hazards

Internal Hazards		
Internal fire	Internal flooding	Missile from turbine building
Pipeline leakage and breakage (including water tanks, pumps and valves)	Internal missile	Flywheel effect of main pumps
Rejection of valves and control rods	Pipe Whip	Internal explosion
Dropped Load	---	---

T-14.3-5 List of External Hazards

External Hazards		
External Natural Hazards		
Air based natural hazards	Ground based natural hazards	Water based natural hazards
A01 Strong wind A02 Tornado A03 High air temperature A04 Low air temperature A05 Extreme air pressure A06 Extreme rain A07 Extreme snow (including snowstorm) A08 Extreme hail A09 Mist A10 White frost A11 Drought	G01 Land rise G02 Soil frost G03 Animals G04 Volcanic phenomena G05 Avalanche G06 Above water landslide G07 External fire G08 Seismic hazards G09 Karsts	W01 Strong water current (underwater erosion) W02 Low water level W03 High water level W04 High water temperature W05 Low water temperature W06 Underwater landslide W07 Surface ice W08 Frazil ice W09 Ice barriers W10 Organic material in water

External Hazards		
External Natural Hazards		
A12 Saltstorm A13 Sandstorm A14 Lightning A15 Meteorite		W11 Corrosion (from salt water) W12 Solid or fluid (non-gaseous) impurities from ship release W13 Chemical release to water W14 Tsunami
Man-made External Events (Off-site accidents)		
M01 Direct impact from ship collision M02 Explosion after transportation accident M03 Chemical release after transportation accident	M04 Explosion outside plant M05 Explosion after pipeline accident M06 Chemical release outside site	M07 Chemical release after pipeline accident M08 Missiles from military activity M09 Excavation work
Man-made External Events (On-site accidents)		
M10 Direct impact of heavy transportation within site M11 Explosion within the site M12 Explosion after pipeline accident within the site	M13 Chemical release within the site M14 Chemical release after pipeline accident within the site M15 Internal fire spreading from other units on the site	M16 Missiles from other units on the site M17 Internal flood and harsh environment spreading from other units on the site M18 Excavation work within the site area
Other External hazards		
M19 Satellite crash M20 Aircraft crash	M21 Magnetic disturbance	M22 Failure of a dam upstream of the plant

14.3.1.2.2 Hazard Screening

In accordance with IAEA SSG-3, Reference [9], consideration of internal fire and internal flooding is mandatory and a detailed PSA has been produced for these hazards. Currently, for HPR1000 (FCG3) a PSA to consider the seismic hazard has not been undertaken and consequently this is not discussed in this sub-chapter. This omission is recognized in sub-chapter 14.5.

The process adopted for screening of hazards is based on:

a) Characterization of the hazard;

b) Assessment of the impact on the plant;

c) Application of the screening criteria (see below) (a different set of criteria were used for co-incident hazards):

1) Criterion 1: The event is of equal or lesser damage potential than the events for which the plant has been designed. This requires an evaluation of plant design bases in order to estimate the resistance of plant structures and systems to a particular external hazard;

2) Criterion 2: The event has a significantly lower mean frequency of occurrence than another event, taking into account the uncertainties in the estimates of both frequencies, and the event could not result in worse consequences than the consequences from the other event;

3) Criterion 3: The event cannot occur close enough to the plant to affect it. This criterion must be applied taking into account the range of magnitudes of the event for the recurrence frequencies of interest;

4) Criterion 4: The event is included in the definition of another event;

5) Criterion 5: The event is slow in developing, and it can be demonstrated that there is sufficient time to eliminate the source of the threat or provide an adequate response.

If a hazard meets a single screening criterion it is screened out.

The hazards retained for the HPR1000 (FCG 3) PSA following the application of the screening criteria were:

a) A01 Strong wind;

b) A02 Tornado;

c) W10 Organic material in water;

d) W12 solid or fluid (non-gaseous) impurities from ship release;

e) G08 Seismic hazards (undertaken by specific analysis and not considered in the analysis of other external hazards).

The approach used to assess the risk contribution arising from these hazards is summarized in the next sub-chapter.

14.3.1.2.3 Hazards Modeling

As indicated in the preceding sub-chapter, detailed modeling was undertaken for internal fire and internal flooding and an alternative simple form of modeling was undertaken for the remaining screened in hazards. This sub-chapter outlines the approach undertaken in each case and provides concluding remarks.

a) Fire PSA Modeling

The Fire PSA undertaken for HPR1000 (FCG3) follows the guidance presented in NUREG/CR-6850, Reference [10]. The general ignition source frequency of NUREG/CR-7114, Reference [11] is used for analyzing of LPSD states of Fire PSA. The scope of HPR1000 (FCG3) internal fire PSA includes power state and LPSD states. The key steps adopted for the Fire PSA are as follow:

1) Plant boundary definition and partitioning;

2) Fire PSA components selection;

3) Fire PSA cable selection;

4) Qualitative screening;

5) Fire-induced risk model;

6) Fire ignition frequencies;

7) Quantitative screening;

8) Scoping fire modeling;

9) Detailed circuit failure analysis;

10) Circuit failure mode likelihood analysis;

11) Detailed fire modeling;

12) Post-fire human reliability analysis;

13) Fire risk quantification;

14) Uncertainty and sensitivity analysis;

15) Fire PSA documentation.

Some of the key issues associated with the above are detailed in the following text.

Division of Fire Compartments

This was an iterative process. For areas with higher risks and already having specific fire zoning schemes, the division of fire compartments was performed by making reference to such fire zoning schemes; for areas with low risks, such as the conventional island and balance of plant (BOP), the division is made with buildings as the basic units. Further sub-division of the initial building zones was undertaken based on the results of initial quantitative assessments.

Identification of Equipment and Cables important for Fire PSA

This was largely based on the criteria outlined in NUREG/CR-6850, Reference [10]. Equipment is identified as important to the PSA its failure would lead to an IE, if it provides a mitigating function or it informs the operator.

Cables that support operation of equipment important for fire PSA are also included.

Estimation of Ignition Frequency Calculations and Sources of Data

The derivation of the ignition frequencies for fire compartments considers both fixed ignition sources in the compartment and temporary ignition sources. Data for power operation is sourced from NUREG/CR-6850, Reference [10] and the data for Shutdown is sourced from NUREG/CR-7114, Reference [11].

Consideration of fire consequences

The modeling of fire impacts is established by considering the various different consequences of the fire. These include the IEs caused by the fire, the availability of equipment affected by the fires, the habitability for operations personnel and the feasibility of measures to mitigate the fire impacts. The approach adopted reflects that outlined in NUREG/CR-6850, Reference [10].

Plant response analysis and its model

The Fire PSA model developed uses the plant faults (internal events) PSA model as the basis and uses the features available in the RiskSpectrum PSA program to reflect the loss/availability of equipment in the zones affected by the fire (and fire spread), the feasibility of assumed operator responses and the effectiveness of fire mitigation (suppression) measures.

Similarly, this information is applied to each POS models associated with the LP SD PSA.

b) Internal Flooding PSA Modeling

The Internal Flooding PSA undertaken for HPR1000 (FCG3) follows the Chinese PSA technical standard NB/T 20037.3, Reference [12] for flooding in nuclear power plants the standard (referred to in sub-chapter 14.3). The scope of HPR1000 (FCG3) internal flooding PSA includes power state and LPSD states. The key steps adopted for the Internal Flooding PSA are as follows:

1) Flooding areas definition;
2) Identification of flooding sources, flooding mechanisms & Structures, Systems and Components (SSC) impacted by flooding;
3) Plant walkdown;
4) Qualitative screening of flooding areas;
5) Characterization of flooding scenarios;
6) Flooding IE frequency quantification;
7) Flooding consequence analysis;

8) Flooding mitigation & human reliability analysis;

9) PSA modeling of flooding scenarios;

10) Quantification of flood-induced accident sequences;

11) Documentation of internal flooding PSA.

The approach adopted for the development of Level 1 PSA model for internal flooding can be divided into two phases, qualitative analysis and quantitative analysis. In the qualitative phase the plant boundary for the analysis is defined and the plant is divided into a number of analysis cells (i.e. flooded areas), information on the equipment and flooding sources is obtained and qualitative screening for the flooded areas is undertaken. In the quantitative phase, detailed flooding scenario analysis, human factors analysis and flooding impact analysis, etc. is undertaken for the screened in flooded areas, flooding event frequency in each flooded area is established and Level 1 PSA model for internal flooding is developed to derive the quantitative results of CDF caused by internal flooding.

Similarly, this information is applied to each of the POS models associated with the LPSD PSA.

As can be seen from the above text, the underlying principles applied to the development of the Internal Flooding PSA are similar to those applied in the Fire PSA. Consequently, specific details supporting the methods and assumptions are not detailed here.

Similarly to the Fire PSA modeling, the Internal Flooding model uses the plant faults (internal events) PSA model as the basis and uses the features available in the RiskSpectrum PSA program to reflect the loss/availability of equipment in the zones affected by the flooding (and flooding spread) and the feasibility of assumed operator responses.

14.3.1.2.4 Analysis of Other External Hazards

The approach adopted for the Level 1 PSA of the other screened in hazards can be considered to comprise three major technical elements:

a) Hazard analysis of external hazards;

b) External hazard fragility analysis;

c) External hazard plant response model.

Envelopment analysis is used to analyze external hazards (single hazards and coincident hazards) with an occurrence frequency below 1.0E-07/ry in a conservative manner. The Core Damage Frequency (CDF) obtained from envelopment analysis is added to the total CDF for external hazards.

For those hazards assessed to occur more frequently than 1.0E-07/ry, a more detailed analysis was performed.

The following hazard was assessed using the envelopment (bounding) approach:

a) A02 – Tornado.

These requiring more detailed analysis were:

a) A01 - Strong wind;

b) W10 Organic material in water;

c) W12 Solid or fluid (non-gaseous) impurities from ship release.

14.3.1.2.5 Sub-Chapter Conclusions

The Level 1 PSA for hazards (external events) has been developed taking due cognizance of recognized standards and provides a strong foundation on which the UK HPR1000 PSA can be developed. It is recognized that the HPR1000 (FCG3) PSA does not currently consider the seismic hazard. The key findings arising from the development of the HPR1000 (FCG3) and areas for consideration for enhancement to support the development of the UK HPR1000 are identified in sub-chapters 14.4 and 14.5 respectively.

14.3.1.3 Fuel Route

In HPR1000 (FCG3) the risk arising from fuel route operations is represented in the SFP PSA. The SFP PSA calculates the Fuel Damage Frequency (FDF) and covers all types of potential IEs and all associated operational modes. The analysis method of the internal event SFP PSA is similar to internal event Level 1 PSA (i.e. use of event tree and fault trees). The key differences associated with the SFP PSA can be summarized as:

a) The consequences to be assessed are:

1) Fuel Damage;

2) Fuel exposure (loss of cooling, or water).

b) Operating states are defined as:

1) Normal operation;

2) Refuelling operation.

c) Hazards are considered to induce the IEs represented in the plant faults (internal events) SFP PSA;

d) In the case of loss of coolant in the SFP, it takes a long time to reach the dangerous water level and such an accident can be promptly identified and handled in routine operational inspections.

The guidance presented in NB/T 20037.1-2011 (referred to in sub-chapter 14.3), allows truncation of event sequences and related system models with the selected low enough cut-off value, so as to avoid neglecting the relevance related to the important minimal cut sets or accident sequences (such as the relevance of personnel errors). Based on the quantitative calculations and comparisons of the PSA model for the SFP, the cut-off value is determined as 1E-13.

14.3.2 Level 2 PSA

The scope of the HPR1000 (FCG3) Level 2 PSA encompasses:

a) Plant states: Power operation and LPSD states;

b) IEs: Plant faults (internal events) and Hazards (external events).

The approach adopted for the HPR1000 (FCG3) PSA is outlined in the remainder of this sub-chapter.

14.3.2.1 Plant Faults (Internal Events) Level 2 PSA

The Level 2 PSA undertaken for HPR1000 (FCG3) follows the guidance presented in IAEA-SSG-4, Reference [13]. The scope of the HPR1000 (FCG3) Level 2 PSA includes power state and LPSD states. The key steps adopted for the Level 2 PSA are as follow:

a) Investigation of plant information and important designs for severe accident;

b) Classification of plant damage states and estimation of their frequencies;

c) Accident progression and containment capacity analysis;

d) Quantification of the Large Release Frequency:

1) Classification of release categories and estimation of their frequencies;

2) Source term analysis for each release category.

e) Uncertainty and sensitivity analysis;

f) Documentation.

Some of the key issues/elements associated with the above are detailed in the following text.

Analysis of Plant Damage States (PDS)

The PDSs are used to link the Level 1 core damage sequences to the Level 2 Containment Event Trees (CET). Core damage sequences with similar characteristics are grouped and linked to the appropriate CET by a linking event tree.

The definition of the plant damage states (PDS) used on HPR1000 (FCG3) is based on the PDS presented in IAEA-SSG-4 , Reference [13]. The PDS used are presented in T-14.3-6.

T-14.3-6 HPR1000 (FCG3) Level 2 PSA Plant Damage States

PDS	Description
---	**Power Operation**
LL	Large/medium LOCA in POSA
SL	Small LOCA in POSA, primary high pressure
IS	Interface LOCA in POSA
TR	POSA transient accident, primary high pressure
VS	POSA main steam/feedwater pipe break accident, primary high pressure
SLD	Small LOCA in POSA, primary low pressure
SG	Core damage caused by SG tube rupture in POSA, primary high pressure
SS	Core damage caused by SLB+SGTR in POSA
VR	Large break in pressure vessel (IE)
AT	Primary high-pressure core melt transient caused by Anticipated Transient Without Scram (ATWS) accident
TP	Primary high-pressure core melt transient caused by station black out accident
---	**Shutdown Operation**
LL_S	Large/medium LOCA in POSB and C
SL_S	Small LOCA in POSB, C and D, primary high pressure
IS_S	Interface LOCA in shutdown condition
TR_S	POSB, C and D transient accident, primary high pressure
VS_S	POSB main steam/feedwater pipe break accident, primary high pressure
SLD_S	Small LOCA in POSB, C, D and G, primary low pressure
SG_S	Core damage caused by SG tube rupture in POSB, primary high pressure
SS_S	Core damage caused by SLB+SGTR in POSB
TP_S	Primary high-pressure core melt transient caused by station black out in POSB, C, D and G
VO_S	Core melt accident in POSE, primary loop already open
VO_S	Core melt accident in POSE, primary loop already open

Analysis of severe accident phenomenon

The analyzed severe accident phenomena include:

a) Induced break in the primary loop;

b) Early-stage overpressure of containment;

c) Primary water injection (core reflooding);

d) Hydrogen phenomena;

e) Steam explosion;

f) Phenomena after pressure vessel failure (pressure vessel up lift, Direct Containment Heating (DCH), etc.);

g) Late-stage overpressure of containment;

h) Molten Core-Concrete Interaction (MCCI).

Analysis of structural performance of containment

The probability of containment failure with respect to pressure function used in the HPR1000 (FCG3) PSA is conservatively based on the analysis for CPR1000 units.

Containment event tree

The Containment Event Trees (CETs) have been developed to reflect, chronologically, the following stages of the accident progression:

a) Before pressure vessel failure (TF1);

b) Pressure vessel failure (TF2) and;

c) Long-term stage after pressure vessel failure (TF3).

The mitigation functions presented in the CETs include:

a) Containment isolation. Auto/manual closure of valves on through-containment pipework to ensure that the fission products will not leak out through such paths;

b) Reactor cavity water injection. Manual activation of the reactor cavity water injection function of the Containment Heat Removal System (EHR [CHRS]);

c) Primary depressurization. Manual activation of the severe accident dedicated valve to avoid high-pressure core melt;

d) Containment heat removal. Realize removal of heat from the containment through the HER [CHRS] system;

e) Passive hydrogen recombiner. To control the hydrogen concentration in the containment in the course of severe accident and reduce the threats of hydrogen combustion and explosion on the integrity of containment.

Quantification of functional titles in event tree

The methods used for the assessment of the failure probabilities of the mitigating systems is based on the use of fault trees (the approach adopted is the same as that used for the Level 1 PSA described under the *System Reliability Analysis* heading in sub-chapter 14.3.1.1). In some cases (e.g. the probability of conditional failure of containment) data from NUREG/CR-6595R1, Reference [14] has been used.

Quantification of the Large Release Frequency (LRF)

The quantification of the Level 2 PSA has been undertaken using the RiskSpectrum PSA program (Version 1.3.2) of Scandpower AB.

14.3.2.2 Internal and External Hazards Level 2 PSA

The Level 2 PSA for hazards is developed based on the Level 1 PSA for external events and the scope of analysis is the same as that of Level 1 PSA. The main risks in power and shutdown conditions come from:

a) Internal fires;

b) Internal flooding;

c) Other external events.

The PDS attributes and characteristic quantities are essentially the same as those for internal events. Other aspects, such as the establishment of CET and the methods to analyze the accident progresses and phenomena, are the same as those for plant faults (internal events) and are not restated here.

14.3.2.3 Sub-Chapter Conclusions

The Level 2 PSA has been developed taking due cognizance of recognized standards and provides a strong foundation on which the UK HPR1000 PSA can be developed. However, it is acknowledged that the HPR1000 (FCG3) PSA does not currently include consideration of source terms.

The key findings arising from the development of the HPR1000 (FCG3) and areas for consideration for enhancement to support the development of the UK HPR1000 are identified in sub-chapters 14.4 and 14.5 respectively.

14.3.3 Level 3 PSA

Level 3 PSA has not been undertaken for HPR1000 (FCG3). The requirement for this in support of the UK HPR1000 is recognized in sub-chapter 14.5.

14.4 HPR1000 (FCG3) PSA Key Findings

This sub-chapter provides a summary of the information available from the HPR1000 (FCG3) PSA and considers the following:

a) The influence that the HPR1000 (FCG3) PSA had on the evolution of the HPR1000 (FCG3) design;

b) The results available from the HPR1000 (FCG3) PSA together with a brief discussion of the key contributors and a summary of the measures identified that have the potential to reduce the assessed risks.

14.4.1 HPR1000 (FCG3) PSA Influence on the HPR1000 (FCG3) Design

The PSA developed through the course of the design of HPR1000 (FCG3), has supported the design evolution through evaluation of:

a) Three series configurations of the safety system;

b) Containment heat removal system;

c) Definition of DEC-A (Design Extension Condition A).

Potential risk reduction measures were also identified for consideration, for example, provision for manual operation of the reactor cavity water injection valves.

14.4.2 HPR1000 (FCG3) PSA Results

This sub-chapter presents a brief summary of the results available from the HPR1000 (FCG3) PSA. Results are presented for:

a) Core Damage Frequency – Level 1 PSA;

b) Large Release Frequency – Level 2 PSA.

14.4.2.1 Level 1 PSA Results

The Level 1 PSA considers Power operations and LPSD operations of the reactor and encompasses both plant faults and a number of internal and external hazards.

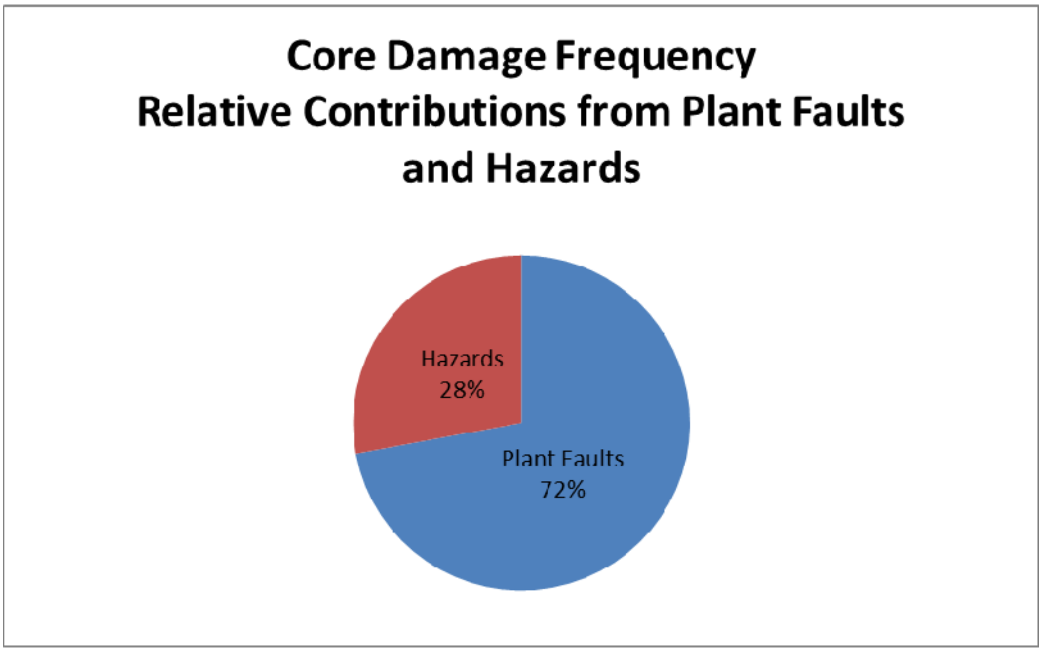
The CDF predicted in the HPR 1000 (FCG3) PSA is 6.49E-07/ry[2]. The contributions associated with plant faults and internal/external hazards[3] are shown in the following table and diagrammatically in F-14.4-1.

T-14.4-1 Core Damage Frequency of Plant Faults and Hazards

Case	Core Damage Frequency (/ry)		
	Plant Faults	**Internal/External Hazards**	**Total**
Reactor (all modes)	4.70E-07	1.79E-07	6.49E-07

[2] Excluding the contribution of Seismic PSA (SPSA), which is still ongoing as the progress of detailed design in HPR1000 (FCG3). According to a conservative assessment on the SPSA, the total CDF meets, with sufficient margins, the design target of HPR1000 (FCG3) as specified in reference to that required in EUR, Reference [15].

[3] Excluding the SPSA as explained in Note 1.

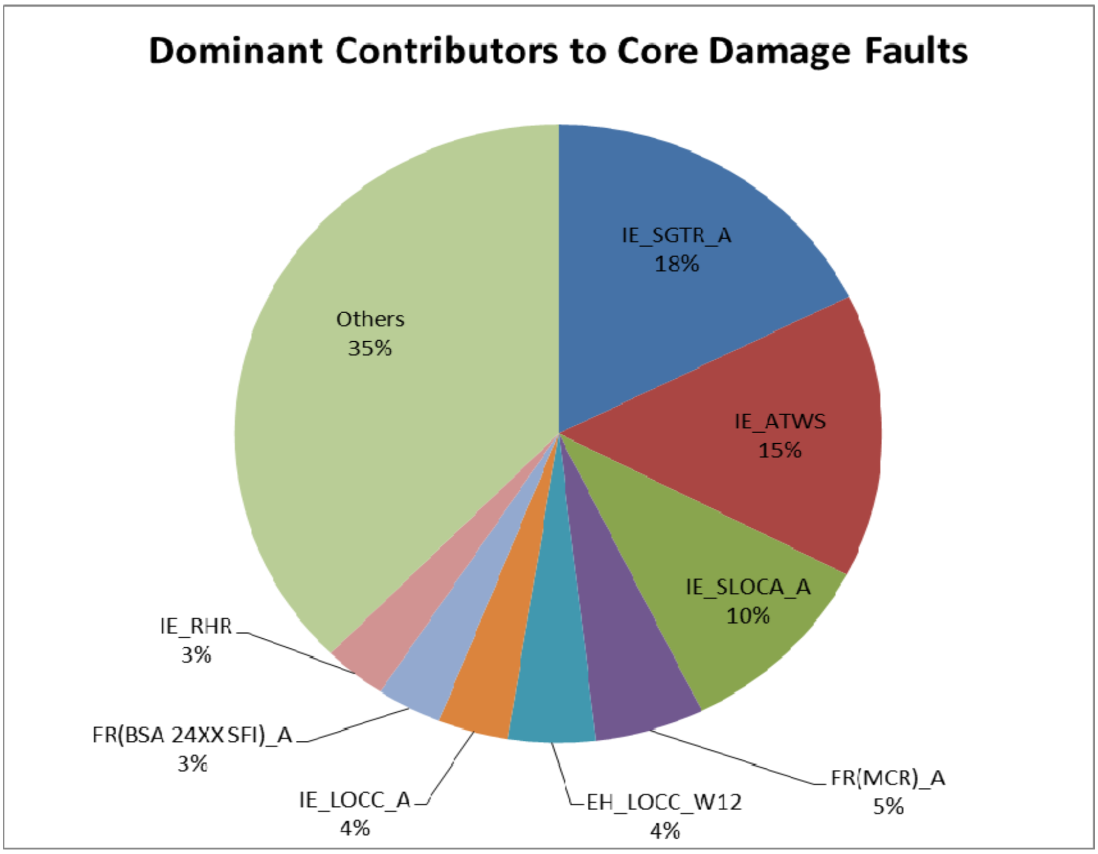


F-14.4-1 CDF % Contribution from Plant Faults and Hazards

The dominant contributors to the C DF are identified in T-14.4-2 and F-14.4-2 shows their % contribution alongside all other contributors.

T-14.4-2 Dominant Contributors (Plant Faults and Hazards) to CDF

No	ID	Description	CDF(/ry)
1.	IE_SGTR_A	IE SGTR in PO	1.18E-07
2.	IE_ATWS	IE ATWS in PO	9.89E-08
3.	IE_SLOCA_A	IE small LOCA in PO	6.78E-08
4.	FR(MCR)_A	Fire in MCR room in PO	3.63E-08
5.	EH_LOCC_A	LOCC induced by external hazard in PO	2.91E-08
6.	IE_LOCC_A	IE LOCC in PO	2.34E-08
7.	FR(BSA 24XX SFI)_A	Fire in Saf eguard Building A cable room at +4.9m in PO	2.17E-08
8.	IE_RHR	IE RHR in PO	2.10E-08



F-14.4-2 Dominant contributors to CDF (Plant Faults and Hazards)

As can be s een from the above, the m ost dominant contributions arise from SG tube rupture accident, ATWS accident and loss of coolant accident caused by primary small break (internal events during power opera tion), accounting for 18%, 15% and 10% of the total CDF respectively. In the case of the SG tube ru pture and th e small break LOCA events, their con tribution reflects a known conservatism in the Human Error Probability modeling associated with feed and bleed. The ATWS event is a dom inant contributor essentially as a result of the fact that there are many events that lead to an ATWS event.

Additionally, it can be seen that the distribution of the contributions from all IEs can be considered to be well balanced.

The risk arising from fuel route operations is represented in the Spent Fuel Pool (SFP) PSA. It calculates the fuel dam age frequency (FDF) and covers all types of potential IEs and all associated o perational modes. The result (FDF) of FCG Unit3 is ab out 3.82E-09/ry.

14.4.2.2 Level 2 PSA Results

The Level 2 PSA considers the output fr om the Le vel 1 PSA together with contributions arising from the fuel route operations.

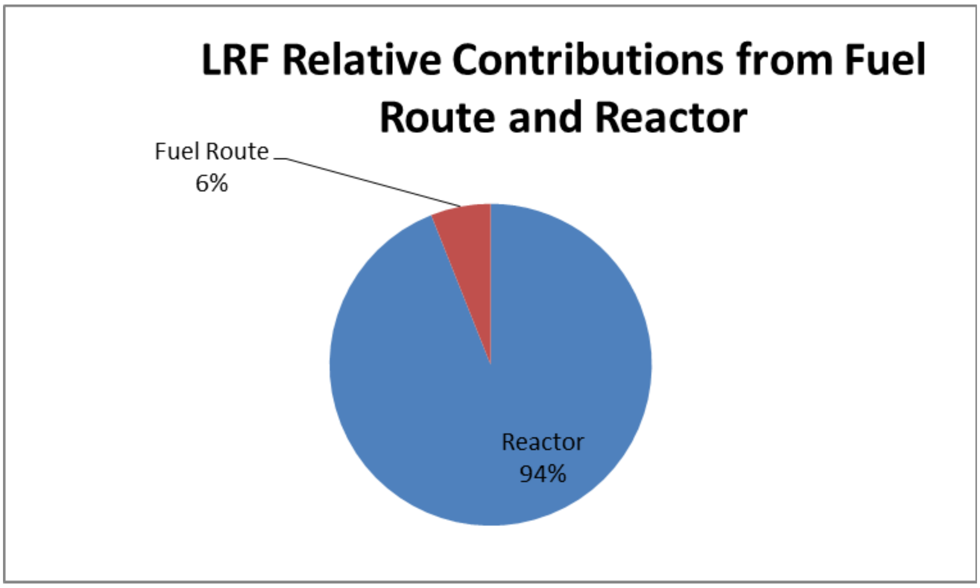
The LRF predicted in the HPR1000 (FCG3) PSA is 6.47E-08/ry[4].

14.4.2.2.1 Facility Contribution to LRF

The relative contributions to the LRF associated with the reactor and fuel route facilities are shown in the following table and diagrammatically in F-14.4-3.

T-14.4-3 Large Release Frequency of Reactor and Fuel Route Facilities

Case	Large Release Frequency (/ry)		
	Reactor	**Fuel Route**	**Total**
Facility	6.09E-08	3.82E-09	6.47E-08



F-14.4-3 LRF % Contribution from Reactor and Fuel Route Facilities

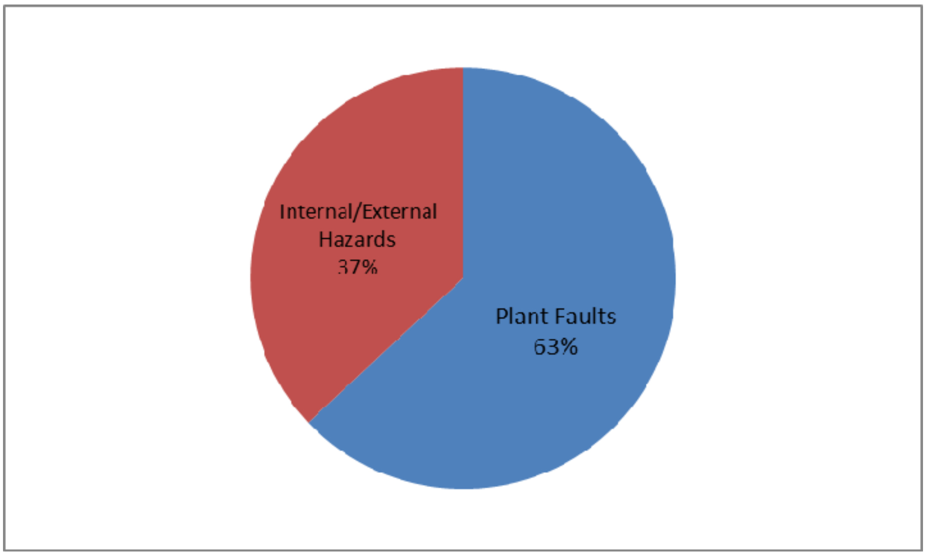
14.4.2.2.2 Plant Faults and Hazards Contribution to LRF

The relative contributions to the LRF associated with plant faults and internal/external hazards are shown in the following table and diagrammatically in F-14.4-4.

T-14.4-4 Large Release Frequency of Plant Faults and Hazards

Case	Large Release Frequency (/ry)		
	Plant Faults	**Internal/External Hazards**	**Total**
Facility LRF	4.07E-08	2.40E-08	6.47E-08

4 Excluding the contribution of Seismic PSA (SPSA), which is still ongoing as the progress of detailed design in HPR1000 (FCG3). According to a conservative assessment on the SPSA, the total LRF meets, with sufficient margins, the design target of HPR1000 (FCG3) as specified in reference to that required in EUR, Reference [15].



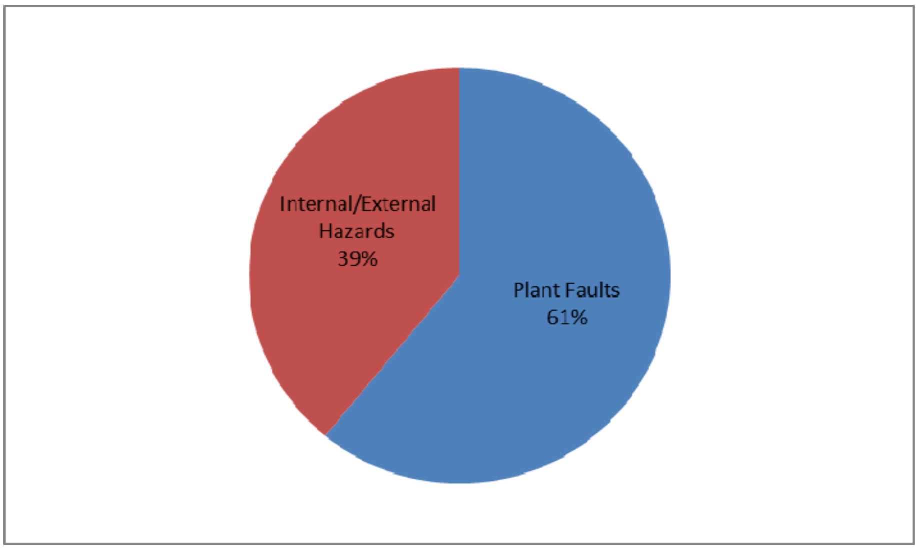
F-14.4-4 LRF % Contribution from Plant Faults and Hazards

14.4.2.2.3 Reactor Contribution to LRF

The reactor's contribution to the LRF split be tween plant faults and interna l/external hazards are shown in the following table and diagrammatically in F-14.4-5.

T-14.4-5 Large Release Frequency of Plant Faults and Hazards

Case	**Large Release Frequency (/ry)**		
	Plant Faults	**Internal/External Hazards**	**Total**
Reactor (all modes)	3.69E-08	2.40E-08	6.09E-08



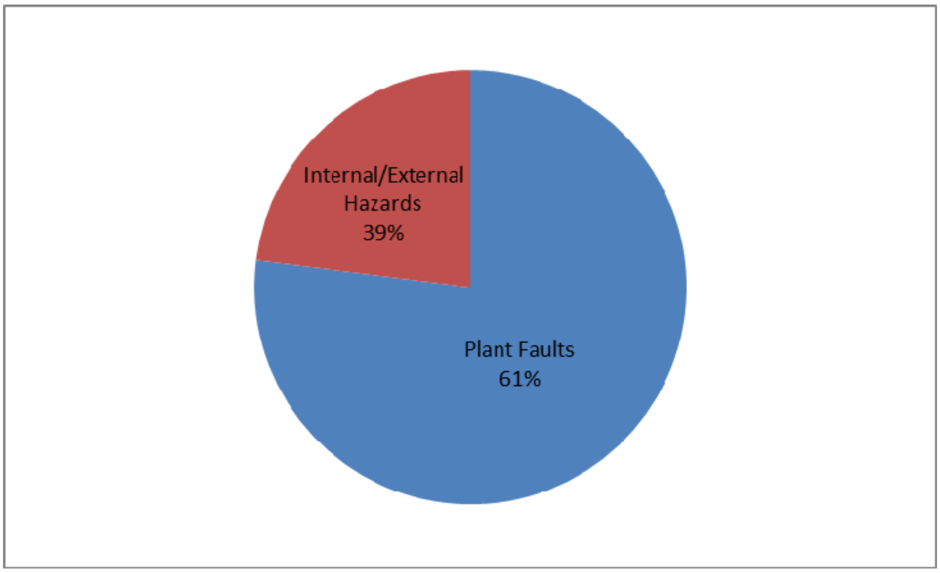
F-14.4-5 Reactor contribution to LRF from Plant Faults and Hazards

14.4.2.2.4 Fuel Route Contribution to LRF

The fuel route's contribution to the LRF split between plant faults and internal/external hazards are shown in the following table and diagrammatically in F-14.4-6.

T-14.4-6 Large Release Frequency of Plant Faults and Hazards

Case	Large Release Frequency (/ry)		
	Plant Faults	Internal/External Hazards	Total
Fuel Route	2.95E-09	8.70E-10	3.82E-09



F-14.4-6 Fuel route contribution to LRF showing ratio between Plant Faults and Hazards

While the HPR1000 (FCG3) PSA has been approved by the Chinese regulator, there are a number of areas where further development work has been agreed. This includes consideration of the release categories associated with the Level 2 PSA. Consequently it is not possible, at this time, to present a breakdown of the LRF across the release categories.

14.5 Development of UK HPR1000 PSA

This sub-chapter outlines the areas of development required to produce a UK HPR1000 PSA that aligns with the regulatory requirements of the UK. This sub-chapter reflects the delta between the methods and information used in the production of the HPR1000 (FCG3) PSA and that required for the UK.

In overall terms, the goal for the UK HPR1000 PSA is, during the course of GDA, to be developed into a full scope PSA (Levels 1, 2 and 3), in line with the design evolution of the UK HPR1000, that considers all operating modes and the full range of fault conditions such that the nuclear safety risks are understood and are ALARP. In supporting of this goal, the assessed risk will be compared with the relevant numerical targets (7, 8 and 9) presented in the Office for Nuclear Regulation (ONR) SAPs.

During the GDA stage, the UK HPR1000 PSA will make some assumptions due to lack of specific information. Assumptions made regarding the behavior of the UK HPR1000 or its operators will be justified and captured. The sensitivity to some important assumptions will be analyzed.

The UK HPR1000 PSA will be developed based on HPR1000 (FCG3) PSA model. The design differences between UK HPR1000 and HPR1000 (FCG3) will be identified and modified in PSA model to ensure it's realistic.

14.5.1 Development of the PSA to Enable Direct Comparison with the Numerical Targets Defined in the ONR SAPs

The HPR1000 (FCG3) PSA derives Core Damage Frequency (Level 1) and Large Release Frequency (Level 2) output.

The UK HPR1000 PSA will be developed through the course of the GDA process to enable direct comparison with the following numerical targets:

Target 7

Individual risk to people off the site from accidents	**Target 7**
The targets for the individual risk of death to a person off the site, from accidents at the site resulting in exposure to ionising radiation, are: BSL: 1×10^{-4} pa BSO: 1×10^{-6} pa	

Target 8

Frequency dose targets for accidents on an individual facility – any person off the site	**Target 8**

The targets for the total predicted frequencies of accidents on an individual facility, which could give doses to a person off the site are:

Effective dose, mSv		**Total predicted frequency per annum**	
		BSL	**BSO**
0.1 – 1	1	1	x 10^{-2}
1 – 10		1 x 10^{-1}	1 x 10^{-3}
10 – 100	1	x 10^{-2}	1 x 10^{-4}
100 – 1000	1	x 10^{-3}	1 x 10^{-5}
> 1000	1	x 10^{-4}	1 x 10^{-6}

Target 9

Total risk of 100 or more fatalities	**Target 9**

The targets for the total risk of 100 o r more fatalities, either immediate or eventual, fro m accidents at the site resulting in exposure to ionising radiation, are:

BSL: 1 x 10^{-5} pa
BSO: 1 x 10^{-7} pa

14.5.2 Fault Schedule

The process adopted for the identification of IEs for the HPR1000 PSA is based on the use of a standardized list of IEs supplem ented by inform ation from other operating plant (see sub-chapter 14.3.1.1).

The need for a suitable dem onstration that the process used to identify all relevant IEs and that these are presented in the form of a Fault Schedule is recognized in chapter 12. The fault schedule produced in line with the guidance presented in ch apter 12 will support development of the UK HPR1000 PSA.

The need for a fault schedule and the intent with respect to its production is presented in chapter 12 of the PSR.

14.5.3 Fuel Route Operations

The scope of operations considered for the HPR1000 (FCG3) Spent Fuel Pool P SA currently considers storage of spent fuel in the spent fuel pool and transfer of spent fuel to/from the reactor.

Expansion of the scope of operations to include consideration of Spent Fuel Interim Storage (SFIS) operations will be considered at an appropriate stage in the licensing process.

14.5.4 Seismic PSA

While it is noted that the HPR1000 (FCG3) PSA has yet to incorporate the risk arising from a seismic hazard, this activity is site-specific and will be assessed at the appropriate stage in the licensing process.

However, as part of de-risking the overall licensing process, consideration will be given to undertaking this work for UK HPR1000 in an earlier stage of the licensing process. In this event generic site characteristics will be used to allow the PSA to establish the risk contribution arising from seismic hazards and to allow insights into the hazard withstand capabilities of the UK HPR1000 design.

14.5.5 Development of the Level 2 PSA

The HPR1000 (FCG3) PSA that forms the basis of the reporting in this chapter generates estimates of the LRF with consideration of release categories or source terms. Associated work is underway to reflect this information in the HPR1000 (FCG3) PSA and hence will be incorporated into the UK HPR1000 PSA at the appropriate time.

14.5.6 Level 3 PSA

14.5.6.1 Requirement

Level 3 PSA will be developed through the GDA process to provide evidence that the UK HPR1000 design will be assessed against the ONR SAPs targets 7 to 9, as well as other ONR expectations related to Level 3 PSA (i.e. Table A1-4 of NS-TAST-GD-030, Reference [16]). Level 3 PSA presents the consequences of a severe accident, or non-core damage radionuclide release at a nuclear facility. These consequences may be probabilistic or deterministic in nature. Probabilistic consequences are the likelihood of long term health consequences to those affected by the radionuclide release. Deterministic consequences are the estimated immediate health effects and economic effects as a result of the radionuclide release.

14.5.6.2 Overview of Level 3 PSA Method

Level 3 PSA starts where Level 2 PSA ends. Level 2 PSA models the containment response to a severe core accident. The conclusion of a Level 2 PSA is the characterization of the radionuclide release including frequencies of the release, quantities of the nuclides and the energy, timing and location of the release. The characterization of these releases is usually grouped by their similarity of effect on the environment. This full characterization of a release is referred to as a source term.

Level 3 PSA uses source term information from the Level 2 PSA to calculate the source term release consequences on the environment. Level 3 PSA computer codes apply an atmospheric transportation and dispersion model to the radionuclide releases to calculate the distribution of the radionuclides geographically. The atmospheric transportation and dispersion model will typically include options to modify the quantity of radionuclides in the dispersion plume for events such as: wet deposition (due to rain), dry deposition, resuspension, etc. These options will remove or add quantities of radionuclides from or into the plume as it is calculated to spread across the region.

Countermeasures and protective actions are generally used in Level 3 PSA computer codes to make the consequences more realistic or less conservative in nature. Typical countermeasures include timing of mandatory evacuation, or sheltering; control of ingestion of contaminated food stuffs, timing for consumption of potassium iodide, etc. These countermeasures will reduce the quantity of radionuclide that comes into contact with humans in the affected zones.

After the atmospheric transportation and dispersion model has predicted where the released radionuclides have settled, Level 3 PSA computer codes use dose calculations to determine the immediate and long-term health effects on humans from the radiation and long term exposure to the contaminated environment. For example, high initial levels of exposure may result in severe short term health effects, while low levels of exposure over a long period of time may result in a long-term health effect. Level 3 PSA computer codes use these results to predict the probability of an individual, at a certain location, risk of death or other serious health effects within a certain time range. The Level 3 PSA codes also use the dispersion model to calculate the economic consequences due to the release.

14.5.6.3 Information Required to Support Level 3 PSA

Level 3 PSA requires several inputs in order for the analysis to be full scope and comprehensive in scale: recent geographical distribution of humans and biota in the target area; source terms, and; meteorological data.

The first input requirement is a recent geographical projection of the inhabitants, both human, farm animals and human food growing in the projected area affected by the accident. This information can be very detailed as humans eat a variety of produce, dairy and meat, and these may all be contaminated in the fallout from the accident. The geographical information should also be able to have enough information that the Level 3 software can estimate the financial cost from the fallout due to loss of business in the affected area, costs of clean-up and moving people to areas outside the zone, etc. Until such time as the specific site and associated site and meteorological, characteristics are defined, generic site characteristics will be used.

The second input requirement is the output from the Level 2 PSA, the source terms. These characterizations of the released radionuclides contain information that is used in Level 3 computer codes to model the atmospheric transportation and dispersion of the released plume of radionuclides.

The third input requirement is a sufficient quantity of meteorological data. The Level 3 PSA software samples this data to build the atmospheric transportation and dispersion models. This process is repeated many times so that a prediction of the consequences is calculated with a high confidence of accuracy.

The Level 3 software will use these three inputs to determine the consequences of the release and hence allow a comparison with the ONR SAPs targets 7, 8 and 9.

14.5.6.4 Potential Software Options for Delivery of a Level 3 PSA

It is recognized that, in the UK today, there are two computer packages used to undertake Level 3 PSA calculations.

The first is an older suite of software, PC-COSYMA. PC-COSYMA is generally accepted for use in Level 3 PSA calculations in the UK. However PC-COSYMA was created some decades ago and is somewhat difficult to run on modern computers. It has most of the functionality that a modern Level 3 code should have, except that the geographical population information is somewhat dated and will not likely be updated to modern information. This is a potential limitation of this software.

PACE is a new suite of software, developed by Public Health England (PHE) that is built on the chassis of the off-the-shelf Graphical Information System (GIS) software, ArcGIS. PACE has the ability to use the state of the art atmospheric transportation and dispersion model NAME3 (a Lagrangian 3D dispersion model), as well as the traditional older atmospheric transportation and dispersion model ADEPT (a Gaussian plume model). NAME3 is considered more realistic than the older methods of modeling dispersion of nuclides as a result of its enhanced functionality, and hence tends to give more realistic results. PACE uses very up-to-date geographical population data that was obtained recently during the last UK census.

CGN is currently considering the software options for delivery of the Level 3 PSA.

14.6 Conclusions arising from the HPR1000 (FCG3) PSA

14.6.1 Assessment Criteria

The PSA for UK HPR1000 will be developed from the Level 1 and Level 2 PSA used in support of the HPR1000 (FCG3) safety analysis and hence this chapter seeks to demonstrate that the PSA for the HPR1000 (FCG3):

a) Was produced based on methods that were consistent with international standards;

b) Has a scope that considers the key operating modes, under the prescribed fault conditions;

c) Produces risk estimates for comparison with numerical targets.

PSA has supported and will continue to support the design/evolution of the HPR1000 (FCG3) to optimize the safety and environmental performance and hence this chapter seeks to demonstrate that the PSA:

a) Has been in use to support the development of the HPR1000 (FCG3) design;

b) Through the process of GDA, will be used as a tool to support current/future design development of the UK HPR1000.

The HPR1000 (FCG3) PSA is intended to demonstrate that the HPR1000 (FCG3) design meets the following overall safety objectives (from EUR (revision D), Reference [15]) as follows:

a) For HPR1000 (FCG3) Level 1 PSA the CDF due to all internal and external events shall be lower than 1E-5/ry;

b) For HPR1000 (FCG3) Level 2 PSA the LRF shall be lower than 1E-6/ry.

It is recognized that an alternative set of targets are defined as applicable to the UK HPR1000 PSA. These, and the need to provide a comparison with these, are noted in the preceding sub-chapter.

14.6.2 Conclusions

Notwithstanding the areas identified for development in sub-chapter 14.5, the following conclusions are drawn.

14.6.2.1 Qualitative Aspect

a) The HPR1000 (FCG3) PSA has been developed using methods that are consistent with international standards (see text in sub-chapter 14.3);

b) The HPR1000 (FCG3) PSA addresses the key facilities and operations (some additional scope is identified in sub-chapter 14.5);

c) There is strong evidence that the PSA has been used to positively support the evolution and optimization of the HPR1000 (FCG3) PSA (see sub-chapter 14.4.1) and that this good practice will continue to support the evolution of the UK HPR1000.

14.6.2.2 Quantitative Aspect

The results presented in sub-chapter 14.4.2 allow the following comparisons:

HPR1000 PSA	Target
Level 1 CDF	< 1E-05/ry
Level 2 LRF	< 1E-06/ry

For both the Level 1 and the Level 2 results there remain a good margin between the assessed risk and the associated CDF/LRF targets.

This margin allows scope for the risk target to continue to be met with the addition of the risk contributions from the enhanced scope identified in sub-chapter 14.5.

14.7 PSA Conclusions

The foundation on which the UK HPR1000 PSA will be developed meets the specific objectives defined to support the high level objectives defined in chapter 1.

The risk estimates generated by the HPR1000 (FCG3) PSA meet the targets defined for CDF and LRF. This provides confidence that the PSA to be developed for UK HPR1000 will be able to demonstrate compliance with the numerical targets defined in the ONR SAPs and that risks are ALARP.

The areas that require enhancement to produce the UK HPR1000 PSA have been identified and these will be incorporated into the UK HPR1000 PSA at appropriate stages in, and beyond, GDA.

14.8 References

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