GE Hitachi Nuclear Energy

NEDO-34187

Revision A

January 2025

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BWRX-300 UK Generic Design Assessment (GDA) Chapter 15.9 – Summary of Results of the Safety Analyses

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EXECUTIVE SUMMARY

This document is Subchapter 15.9, Summary of the Results of the Safety Analyses, of the Preliminary Safety Report of the GEH BWRX-300 for the purposes of UK GDA. It presents the results of the safety analyses which appears in Chapter 15.

ACRONYMS AND ABBREVIATIONS

Acronym	Explanation
ABWR	Advanced Boiling Water Reactor
ALARP	As Low As Reasonably Practicable
AOO	Anticipated Operator Occurrence
ASME	American Society of Mechanical Engineers
BDBA	Beyond Design Basis Analysis
BIS	Boron Injection System
BOC	Beginning of Cycle
BPVC	Boiler and Pressure Vessel Code
BWR	Boiling Water Reactor
CAE	Claims, Arguments and Evidence
CCF	Common Cause Failure
CDF	Core Damage Frequency
CN	Conservative
D-in-D	Defence-in-Depth
DBA	Design Basis Accidents
DEC	Design Extension Condition
DL4	Defence Line 4
DSA	Deterministic Safety Analysis
EOR	End of Rated Cycle
FAP	Forward Action Plan
FDF	Fuel Damage Frequency
FMCRD	Fine Motion Control Rod Drives
FSF	Fundamental Safety Function
GDA	Generic Design Assessment
GEH	GE Hitachi Nuclear Energy
HCU	Hydraulic Control Unit
IAEA	International Atomic Energy Agency
ICS	Isolation Condenser System
ISLOCA	Interfacing System Loss of Cooling Accident
LFWH	Loss of Feedwater Heating
LOCA	Loss of Coolant Accident
LOCV	Loss of Condenser Vacuum
LOPP	Loss of Preferred Power
LRF	Large Release Frequency
LR-TT	Load Rejection-Turbine Trip
MOC	Middle of Cycle

Acronym	Explanation
MSRIVC	Main Steam Reactor Isolation Valve Closure
NBR	Neutron Balance Ratio
ONR	Office of Nuclear Regulation
OPEX	Operational Experience
PI	Pressure Increase
PSA	Probabilistic Safety Assessment
PSR	Preliminary Safety Report
RCPB	Reactor Coolant Pressure Boundary
RGP	Relevant Good Practice
RIV	Reactor Isolation Valve
RPV	Reactor Pressure Vessel
SA	Severe Accident
SSCs	Structures, Systems and Components
UK	United Kingdom
USNRC	U.S. Nuclear Regulatory Commission
ΔCPR/ICPR	Delta Critical Power Range over Initial Critical Power Ratio

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REVISION SUMMARY

Revision #	Section Modified	Revision Summary
A	All	Initial Issuance

15.9 SUMMARY OF RESULTS OF THE SAFETY ANALYSES

Chapter Route Map

This sub-chapter is part of Preliminary Safety Report (PSR) Chapter 15 which presents the BWRX-300 Safety Analyses and comprises the following subchapters:

- PSR Ch. 15.1 – NEDC-34179P, "BWRX-300 UK GDA Ch. 15.1: Safety Analysis - General Considerations," (Reference 15.9-1)
- PSR Ch. 15.2 – NEDC-34180P, "BWRX-300 UK GDA Ch. 15.2: Safety Analysis - Identification, Categorisation, and Grouping of Postulated Initiating Events and Accident Scenarios," (Reference 15.9-2)
- PSR Ch. 15.3 – NEDC-34181P, "BWRX-300 UK GDA Ch. 15.3: Safety Analysis - Objectives and Acceptance Criteria," (Reference 15.9-3)
- PSR Ch. 15.4 – NEDC-34182P, "BWRX-300 UK GDA Ch. 15.4: Safety Analysis - Human Actions," (Reference 15.9-4)
- PSR Ch. 15.5 – NEDC-34183P, "BWRX-300 UK GDA Ch. 15.5: Deterministic Safety Analysis," (Reference 15.9-5)
- PSR Ch. 15.6 – NEDC-34184P, "BWRX-300 UK GDA Ch. 15.6: Probabilistic Safety Assessment," (Reference 15.9-6)
- PSR Ch. 15.7 – NEDC-34185P, "BWRX-300 UK GDA Ch. 15.7: Deterministic Safety Analyses - Analysis of Internal Hazards," (Reference 15.9-7)
- PSR Ch. 15.8 – NEDC-34186P, "BWRX-300 UK GDA Ch. 15.8: Analysis of External Hazards," (Reference 15.9-8)
- PSR Ch. 15.9 – Summary of Results of the Safety Analyses [*this subchapter*]

This layout mainly follows the structure set out in the International Atomic Energy Agency (IAEA) specific safety guide SSG-61 "Format and Content of the Safety Analysis Report for Nuclear Power Plants," (Reference 15.9-9), with the exception that internal and external hazards are discussed in NEDC-34185P (Reference 15.9-7) and NEDC-34196P (Reference 15.9-8).

Sub-Chapter Structure

This sub-chapter presents the acceptance criteria and safety objectives for the BWRX-300 Safety Analyses and comprises the following main sections as in line IAEA SSG-61 (Reference 15.9-9):

- 15.9 – Summary of Results of the Safety Analyses [*this subchapter*]
- 15.9.1 – Results of Analysis of Normal Operation
- 15.9.2 – Results of Analysis of Anticipated Operational Occurrences and Design Basis Accidents
- 15.9.3 – Results of Analysis of Design Extension Conditions without Significant Fuel Degradation
- 15.9.4 – Results of Analysis of Design Extension Conditions with Core Melting
- 15.9.5 – Results of Analysis of Postulated Initiating Events and Accident Scenarios Associated with the Spent Fuel Pool
- 15.9.6 – Results of Analysis of Fuel Handling Events, and Radioactive Releases from a Subsystem or a Component

- 15.9.7 – Results of Analysis of Internal Hazards and External Hazards
- 15.9.8 – Results of Probabilistic Safety Assessment
- Appendix A Claims, Arguments and Evidence
- Appendix B Forward Action Plan (FAP)
- Appendix C Practical Elimination Claims and Provisions
- Appendix D Complementary Defence Line 4 (DL4) Functions for Mitigating Design Extension Conditions
- Appendix E Approach to Development of the Fault Schedule

The Deterministic Safety Analysis (DSA) results for the bounding BWRX-300 events, which meet the acceptance criteria in Tables 15.3-1, 15.3-2, and 15.3-3 of NEDC-34181P (Reference 15.9-3), are provided in the following tables:

- Table 15.9-1: Results Summary of Anticipated Operator Occurrence (AOO) Events
- Table 15.9-2: Results Summary of DBA and Design Extension Condition (DEC) Events – Non-Loss-of-Coolant Accident (LOCA)
- Table 15.9-3: Results Summary of Design Basis Accident (DBA) Events – LOCA

The level 1 and level 2 Probabilistic Safety Assessment (PSA) results for BWRX-300 events are provided in the following tables:

- Table 15.9-4: Core Damage Frequency (CDF)
- Table 15.9-5: Large Release Frequency (LRF)

The results for the MODES 2-6 DSA are not provided at PSR and will be evaluated as part of future work activities.

Implementation of the Defence-in-Depth (D-in-D) concept ensures multiple, independent layers of protection against unacceptable radiation releases. None of the bounding AOOs, DBAs, or DEC Events Without Core Damage analysed approach the USNRC limits for radioactive releases. Consideration of UK requirements will be complete in future work activities.

Interfaces with Other Chapters

This sub-chapter interfaces with the following PSR Chapters:

- PSR Ch. 15 – Safety Analysis Analyses (all other sub-chapters)

Purpose

The purpose of this subchapter is to present the results of the DSA and the PSA for the BWRX-300.

Scope

The scope of this subchapter comprises the safety analyses presented in NEDC-34183P (Reference 15.9-5) and NEDC-34184P (Reference 15.9-6).

Country Specific Material – United Kingdom Step 2 GDA

The PSR is being submitted as part of Step 2 of Generic Design Assessment (GDA) by the United Kingdom (UK) Office for Nuclear Regulation (ONR). GDA is an up-front, non-site-specific assessment of a generic nuclear power plant design. It is intended to determine whether a proposed reactor type could be constructed, operated, and decommissioned in Great Britain. Step 2 is a fundamental assessment of the generic safety, security, and environment protection cases. It is intended to identify potential showstoppers that may preclude deployment of the design.

Specific nuclear safety requirements differ between countries. Significant unique aspects of the UK regulatory regime are the overriding requirement to demonstrate that risks have been managed and reduced to As Low As Reasonably Practicable (ALARP) and clarity on the Claims, Arguments and Evidence (CAE) employed in the safety argument. The demonstration of ALARP is typically achieved through the application of Relevant Good Practice (RGP).

However, these aspects are not directly relevant to this subchapter. Other UK specific aspects are handled through the identification of future work.

Future Work

The need for future work has been identified during the production of PSR Ch. 15. This arises principally for the following reasons:

- Continuing design development
- Development of the preliminary safety arguments
- Country-specific requirements

PSR Subchapter 15.9 directly supports forward actions as described in Appendix B FAP. Each FAP item comprises a concise description of the required work along with a project phase for when it is needed by; an outline of the reason for raising the FAP is also presented.

15.9.1 Results of Analysis of Normal Operation

The analysis of normal operations for stability is described in NEDC-34269P, "Thermal Hydraulics Summary Report," (Reference 15.9-10) and Section 15.5.2 of NEDC-34183P (Reference 15.9-5).

15.9.2 Results of Analysis of Anticipated Operational Occurrences and Design Basis Accidents

The analysis of AOOs is described in Section 15.5.3 of NEDC-34183P (Reference 15.9-5).

The resulting maximum neutron flux, maximum dome pressure, maximum Reactor Pressure Vessel (RPV) bottom pressure, maximum simulated thermal power, and Delta Critical Power Range over Initial Critical Power Ratio (ΔCPR/ICPR) for AOOs are provided in Table 15.9-1.

The analysis of DBAs is described in Section 15.5.4 of NEDC-34183P (Reference 15.9-5).

The summary results of maximum neutron flux, maximum dome pressure, maximum RPV bottom pressure, maximum simulated thermal power, and peak clad temperature from non-LOCA DBAs are provided in Table 15.9-2.

The LOCA DBA summary results for peak cladding temperature, peak containment pressure, and peak containment shell temperature are provided in Table 15.9-3.

15.9.3 Results of Analysis of Design Extension Conditions without Significant Fuel Degradation

The analysis of DEC events without core damage is described in Section 15.5.5 of NEDC-34183P (Reference 15.9-5). The summary results of maximum neutron flux, maximum dome pressure, maximum RPV bottom pressure, maximum simulated thermal power, and peak clad temperature from non-LOCA DECs are provided in Table 15.9-2.

15.9.4 Results of Analysis of Design Extension Conditions with Core Melting

The analysis description of those DEC events associated with core damage is currently addressed in the Level 2 PSA described in NEDC-34184P (Reference 15.9-6).

An outline of the proposed approach to Severe Accidents (SAs) is presented in NEDC-34183P (Reference 15.9-5). FAP PSR15-4 applies to this section as described in Appendix B FAP.

15.9.5 Results of Analysis of Postulated Initiating Events and Accident Scenarios Associated with the Fuel Pool

The analysis of postulated initiating events and accident scenarios associated with the fuel pool is currently addressed in the Level 1 PSA described in NEDC-34184P (Reference 15.9-6).

The deterministic analysis of these events will be presented in NEDC-34183P (Reference 15.9-5) in a future version of this safety case. FAP PSR15.5-30 applies as described in Appendix B FAP.

15.9.6 Results of Analysis of Fuel Handling Events, and Radioactive Releases from a Subsystem or a Component

The criteria used to judge the acceptability of the residual radiological consequences are often specific to different countries and regulatory regimes and will have to be shown to be applicable to the relevant site. Site-specific assumptions (e.g., atmospheric dispersion factors) will have to be addressed for the relevant site. The assumptions behind radiological calculations (e.g., radiation concentrations in reactor coolant and steam) are often specific to different countries and regulatory regimes and will have to be shown to be applicable to the relevant site. The criteria used to judge the acceptability of the residual radiological consequences are often specific to different countries and regulatory regimes and will have to be shown to be applicable to the relevant site.

Therefore, specific calculations for a UK site will therefore need to be presented in a future version of the safety case, including consideration of relevant UK criteria and assumptions. FAP items PSR15.5-31 and PSR15.5-32 pertain.

15.9.7 Results of Analysis of Internal and External Hazards

The results for the analysis of the internal and external hazards are contained in NEDC-34185P (Reference 15.9-7) and NEDC-34186P (Reference 15.9-8).

15.9.8 Results of Probabilistic Safety Assessment

The Level 1 and Level 2 PSA are described in NEDC-34184P (Reference 15.9-6. The general PSA approach and the insights and applications are also described in NEDC-34184P (Reference 15.9-6).

Preliminary PSA results for Core Damage Frequency (CDF) and LRF are provided in Table 15.9-4 and Table 15.9-5, respectively.

Final CDF and LRF results will be presented in this chapter in a future version of this safety case.

Table 15.9-1: Results Summary of AOO Events

Description	Exposure	Max. Neutron Flux, % Neutron Balance Ratio (NBR)	Max. Dome Pressure, MPa (psia)	Max. Vessel Bottom Pressure, MPa (psia)	Max. Simulated Thermal Power, % NBR	ΔCPR/ ICPR
Decrease in Core Coolant Temperature AOOs						
Loss of Feedwater Heating (LFWH) AOO (Subsection 15.5.3.1.1)	Middle of Cycle (MOC)	101.1	7.17 (1040.1)	7.33 (1062.6)	100.4	0.0341
Pressure Increase AOOs						
Load Rejection-Turbine Trip (LR-TT) AOO (Subsection 15.5.3.2.1)	Beginning of Cycle (BOC)	100.0	7.55 (1094.6)	7.70 (1116.3)	100.0	0.0583
1 Main Steam Reactor Isolation Valve Closure (MSRIVC) AOO (Subsection 15.5.3.2.2)	End of Rated Cycle (EOR)	110.7	7.47 (1083.2)	7.61 (1103.1)	100.3	0.0631
Loss of Condenser Vacuum (LOCV) AOO (Subsection 15.5.3.2.3)	BOC	100.0	7.55 (1094.6)	7.70 (1116.3)	100.0	0.0583
Loss of Preferred Power (LOPP) AOO (Subsection 15.5.3.2.4)	BOC	100.0	7.55 (1094.6)	7.70 (1116.2)	100.0	0.0495
Inventory Reduction AOOs						
Feedwater Pump Trip AOO (Subsection 15.5.3.3.1)	EOR	100.0	7.17 (1039.7)	7.32 (1062.1)	100.0	0.0086

Description	Exposure	Max. Neutron Flux, % Neutron Balance Ratio (NBR)	Max. Dome Pressure, MPa (psia)	Max. Vessel Bottom Pressure, MPa (psia)	Max. Simulated Thermal Power, % NBR	ΔCPR/ ICPR
Increase in Reactor Coolant Inventory AOOs						
Inadvertent Isolation Condenser Initiation - One Train (Subsection 15.5.3.4.1)	MOC	116.8	7.17 (1039.7)	7.32 (1062.1)	100.6	0.0464

Table 15.9-2: Results Summary of DBA and DEC Events – Non-LOCA

Description	Exposure	Max. Neutron Flux, % NBR	Max. Dome Pressure, MPa (psia)	Max. Vessel Bottom Pressure, MPa (psia)	Max. Simulated Thermal Power, % NBR	Peak Clad Temperature °C (°F)
Decrease in Reactor Coolant Temperature Event DBA						
LFWH DBA (Subsection 15.5.4.1.1)	MOC	119.1	7.17 (1039.7)[1]	7.32 (1062.1)[1]	115.2	308.0 (586.3)
Increase in Reactor Pressure Events DBAs						
LR-TT DBA[2] (Subsection 15.5.4.2.1)	MOC	544.3	8.69 (1259.8)	8.86 (1285.1)	111.8	511.8 (953.3)
LOPP DBA (Subsection 15.5.4.2.2)	EOR	151.0	8.61 (1249.2)	8.73 (1266.1)	103.5	312.4 (594.3)
RPV Pressure Control Downscale DBA (Subsection 15.5.4.2.3)	EOR	151.4	8.70 (1262.3)[1]	8.89 (1288.9)[1]	103.5	312.6 (594.7)
MSRIVC-Feedwater Isolation Valve DBA (Subsection 15.5.4.2.4)	EOR	158.9	8.61 (1249.4)	8.73 (1266.3)	103.7	312.7 (594.9)
Increase in Reactor Coolant Inventory DBAs						
Feedwater Flow Increase – All Pumps (Subsection 15.5.4.3.1)	MOC	123.3	7.93 (1150.0)[1]	8.09 (1173.6)[1]	115.5	315.5 (599.9)
Inadvertent Isolation Condenser Initiation –All Trains	MOC	114.3	7.17 (1039.7)	7.32 (1062.1)	100.0	308.0 (586.3)

Description	Exposure	Max. Neutron Flux, % NBR	Max. Dome Pressure, MPa (psia)	Max. Vessel Bottom Pressure, MPa (psia)	Max. Simulated Thermal Power, % NBR	Peak Clad Temperature °C (°F)
(Subsection 15.5.4.3.2)						
Decrease in Reactor Coolant Inventory DBAs						
Loss of Feedwater Flow DBA (Subsection 15.5.4.4.1)	MOC	100.0	7.17 (1039.7)	7.32 (1062.1)	100.0	308.0 (586.3)
RPV Pressure Control Open DBA (Subsection 15.5.4.4.2)	MOC	100.0	7.17 (1039.7)[1]	7.32 (1062.1)[1]	100.0	302.8 (577.0)
Analysis of Design Extension Conditions Without Core Damage						
Pressure Increase DECs						
1 MSRIVC DEC (Subsection 15.5.5.2.1)	EOR	208.9	11.19 (1623.2)	11.33 (1643.3)	130.1	727.1 (1340.8)
Complex Sequence Load Rejection DEC (Subsection 15.5.5.2.2)	EOR	118.4	8.03 (1164.3)	8.17 (1184.5)	100.3	309.2 (588.6)
LOCV DEC (Subsection 15.5.5.2.3)	EOR	245.0	9.98 (1447.3)	10.12 (1467.5)	124.5	332.9 (631.2)
LOPP DEC (Subsection 15.5.5.2.4)	EOR	250.1	11.14 (1615.6)	11.28 (1635.5)	130.4	328.9 (624.0)
Reactivity and Power Distribution Anomalies – DECs						
Common Cause Failure (CCF) All Control Rod Withdrawal at Power	MOC	123.8	7.52 (1091.0)	7.68 (1113.3)	115.4	314.0 (597.2)

Description	Exposure	Max. Neutron Flux, % NBR	Max. Dome Pressure, MPa (psia)	Max. Vessel Bottom Pressure, MPa (psia)	Max. Simulated Thermal Power, % NBR	Peak Clad Temperature °C (°F)
(Subsection 15.5.5.3.1)						
Inadvertent Single Control Rod Withdrawal at Power (Subsection 15.5.5.3.2)	BOC	111.8	7.23 (1048.5)	7.38 (1070.9)	111.8	307.5 (585.5)
Decrease in Reactor Coolant Inventory – DECs						
Feedwater Isolation DEC (Subsection 15.5.5.4.1)	MOC	100.0	8.40 (1218.5)	8.50 (1232.1)	100.0	308.0 (586.3)

Note:

1. The simulation of this event is ended before the RPV Pressure Increase (PI) resulted in an Isolation Condenser System (ICS) train initiation.
2. Results are from a bounding case with combined conservatisms.

Table 15.9-3: Results Summary of DBA Events – LOCA

Parameter	Parameter Value
Results for Main Steam Pipe Break Inside Containment, Conservative Case (PSR Ch. 15, Section 15.5)	
Peak cladding temperature	Less than normal operating temperature
Peak containment pressure	423 kPa
Peak containment shell temperature	134 °C
Results for Feedwater Pipe Break Inside Containment, Conservative Case (PSR Ch. 15, Section 15.5)	
Peak cladding temperature	Less than normal operating temperature
Peak containment pressure	407 kPa
Peak containment shell temperature	134 °C
Results for Small Steam Pipe Break Inside Containment, Conservative Case (PSR Ch. 15, Section 15.5)	
Peak Cladding Temperature	Less than normal operating temperature
Peak containment pressure	191 kPa
Peak containment shell temperature	125 °C
Results for Small Liquid Pipe Break Inside Containment, Conservative Case (PSR Ch. 15, Section 15.5)	
Peak cladding temperature	Less than normal operating temperature
Peak containment pressure	191 kPa
Peak containment shell temperature	110 °C

Table 15.9-4: Core Damage Frequency

PSA Events	Core Damage Frequency (/yr)
Internal Events at Power	1.06E-08
Internal Events Low Power & Shutdown[1]	1.20E-09
Seismic Events	8.20E-07
Fire Events	1.50E-08
Hurricane	2.27E-09
Straight Wind	5.57E-10
Tornado	1.29E-09
Internal Flood	5.26E-09
Fuel and Heavy Load Movements	Fuel Damage Frequency (FDF) at Power 2.3E-09 FDF at Low Power Shutdown 1.8E-09
Fuel Pool Events	FDF at 1.3E-08
Total CDF	**8.56E-07**
Total FDF	**1.71E-08**

Note:

1. Including load drop CDF.

Table 15.9-5: Large Release Frequency

Level 2 PSA Scope	Large Release Frequency (LRF) (/yr)
Internal Events at Power	
Internal Events Low Power & Shutdown[1]	1.20E-09
Seismic Events	8.20E-07
Fire Events	2.67E-09
Hurricane	4.73E-11
Straight Wind	1.10E-11
Tornado	7.06E-10
Internal Flood	5.26E-09
Fuel and Heavy Load Movements	4.10E-09
Fuel Pool Events	1.30E-08
Total	**8.49E-07**

Note:

1. Including load drop CDF.

15.9.9 References

15.9-1 NEDC-34179P, "BWRX-300 UK GDA Ch. 15.1: Safety Analysis- General Considerations," Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

15.9-2 NEDC-34180P, "BWRX-300 UK GDA Ch. 15.2: Safety Analysis - Identification, Categorisation, and Grouping of Postulated Initiating Events and Accident Scenarios," Revision 0, GE-Hitachi Nuclear Energy, October 2024.

15.9-3 NEDC-34181P, "BWRX-300 UK GDA Ch. 15.3: Safety Analysis - Objectives and Acceptance Criteria," Rev A, GE-Hitachi Nuclear Energy, Americas, LLC

15.9-4 NEDC-34182P, "BWRX-300 UK GDA Ch. 15.4: Safety Analysis - Human Actions," Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

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15.9-7 NEDC-34185P, "BWRX-300 UK GDA Preliminary Safety Report: Ch. 15.7 – Deterministic Safety Analyses – Analysis of Internal Hazards

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15.9-9 IAEA, SSG-61, Format and Content of the Safety Analysis Report for Nuclear Power Plants. Specific Safety Guide Series, September 2021.

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15.9-11 ONR, ONR SAPs, "Office for Nuclear Regulation Safety Assessment Principles," 2014 Edition.

15.9-12 NEDC-34140P, "BWRX-300 UK GDA Safety Case Development Strategy," GE-Hitachi Nuclear Energy, Americas, LLC.

15.9-13 NEDC-34274P, "BWRX-300 UK GDA Forward Action Plan Register," GE-Hitachi Nuclear Energy, Americas, LLC.

APPENDIX A CLAIMS, ARGUMENTS AND EVIDENCE

A.1 Claims, Arguments and Evidence

The ONR SAPs 2014, “Office for Nuclear Regulation Safety Assessment Principles,” (Reference 15.9-11) identify ONR’s expectation that a safety case should clearly set out the trail from safety claims, through arguments to evidence. The CAE approach can be explained as follows:

- Claims (assertions) are statements that indicate why a facility is safe
- Arguments (reasoning) explain the approaches to satisfying the claims
- Evidence (facts) supports and forms the basis (justification) of the arguments

The GDA CAE structure is defined within NEDC-34140P, “BWRX-300 UK GDA Safety Case Development Strategy,” (Reference15.9-12) and is a logical breakdown of an overall claim that:

“The BWRX-300 is capable of being constructed, operated and decommissioned in accordance with the standards of environmental, safety, security and safeguard protection required in the UK”.

This overall claim is broken down into Level 1 claims relating to environment, safety, security, and safeguards, which are then broken down again into Level 2 area related sub-claims and then finally into Level 3 (chapter level) sub-claims.

A.2 Risk Reduction As Low As Reasonably Practicable

It is important to note that nuclear safety risks cannot be demonstrated to have been reduced As Low As Reasonably Practicable (ALARP) within the scope of a 2-Step GDA.

ALARP Introduction and Approach

The BWRX-300 ALARP position will be based upon four fundamental aspects, which will be incorporated into the BWRX-300 PSR chapters:

- RGP – The application of recognised codes and standards ensures the application of good engineering practice across the design
- Operational Experience (OPEX) – demonstration that international OPEX has been taken into account in the overall design philosophy and in specific system designs
- Optioneering – Key and fundamental design choices should include consideration of a number of options (or collections of options) to identify the most reasonably practicable. All reasonably practicable options to reduce risk should be implemented
- Risk assessment (PSA compliance and balanced design) – The safety analysis presents the totality of the risk assessment including deterministic and DSA/PSA. The chapters listed above do not support compliance demonstration with the quantitative risk ALARP aspects.

BWRX-300 Operational Experience

This section applies to all chapters and describes the origin of the Boiling Water Reactor (BWR) OPEX that has informed the BWRX-300 design. There have been 115 BWRs built and operated around the world with two Advanced Boiling Water Reactors (ABWRs) currently under construction. Currently there are 63 BWRs operational worldwide. The highest concentration being in the USA where 31 of the 94 operating reactors in the country are BWRs. Many of which are among the best operating plants in the world, performing in the ‘best in class’ category. Therefore, the BWRX-300 design approach leverages nine previous generations of GEH BWR technology with greater than 2000 reactor years of operating experience.

The BWRX-300 is the 10th generation BWR and draws heavily from previous designs with additional simplifications and improvements. The BWRX-300 utilises proven in-use materials, off-the-shelf components, design pressures and temperatures drawn from previous experience and the U.S. Nuclear Regulatory Commission approved Economic Simplified Boiling Water Reactor design. GEH also administers and coordinates a BWR Owners' Group which deals with fleet-wide issues, concerns and OPEX. Previous GEH BWR designs have been licensed worldwide, including in the US, Japan, UK, Taiwan, Switzerland, Italy and Spain.

BWRX-300 Key Design Features

The key design development advantages of the BWRX-300, which support risk reduction, are:

- Reduced LOCA risk e.g., inclusion of Reactor Isolation Valves (RIVs), removal of safety relief valves
- Inherent (passive) safety design over historic active design
- Simplified design supporting increased reliability e.g., reduction in the number of components and pipework lengths
- Flexible energy generation, including combined heat and power and hydrogen production capabilities
- Reduced requirement for operator control or intervention
- Modularisation with constructability integrated into the design
- Reduced external event risk from optimised site layout and plant structural integrity
- Designed in accordance with internationally accepted codes, standards, and guidance to support international deployment with minimum changes
- Design pedigree / heritage from previous BWR designs, e.g., UK ABWR
- Use of many proven technologies, e.g., fuel, core design, steam separator system that reduce fuel leakage and subsequent onsite radiation exposures.
- Ability to use non-safety classified and commercial of the shelf equipment in some areas (e.g., BOP).
- Reduced dependence on AC power
- Reduced challenges to safety systems by improved capacity factors
- Reduced shutdown risk because of large reactor coolant system and equipment and ICS pool volumes
- Reduced susceptibility to thermal-hydraulic flow instabilities from previous designs
- Improved control room design, including human factors
- Design reliability program that identifies and targets equipment needed to support plant safety

APPENDIX B FORWARD ACTION PLAN

This chapter supports the following forward actions, as defined within the Forward Action Plan (FAP), NEDC-34274P, "BWRX-300 UK GDA Forward Action Plan Register," (Reference 15.9-13).

Table B-1: Forward Actions Chapter 15.9 Supports

FAP No.	Finding	Forward Actions	Delivery Phase
PSR15.5-28	**Development of Fault Schedule** The provision of a fault schedule, a tabular summary of the essential parts of a nuclear facility's safety case, is RGP in the UK. Although the current Fault List has many features in common with a fault schedule, it does not meet all of the expectations for one. The associated Safety Assessment Principles (SAPs) are ESS.11, and FA.8.	Develop a format for a fault schedule which will meet UK fault schedule expectations whilst reflecting the BWRX-300 engineering and operational philosophy, and the safety case. Utilize the developing fault schedule during the ongoing design development work. The following aspects should be considered in particular: • Bounding faults • Initiating fault frequencies • Unmitigated consequences • Claimed safety measures Consider use of the UK ABWR fault schedule as a starting point.	PCSR/Detailed Design
PSR15.5-30	The list of faults considered in PSR Ch. 15.5 are principally bounding reactor faults during power operation that were selected deterministically early in the design process. UK RGP is for the list of faults to be identified, and justified, in the safety case. The associated Safety Assessment Principles (SAPs) are: FA.2, FA.5, FA.6, EMC.3, EHA.1, and Numerical Target 4.	A systematic and auditable process [eg FMEA, HAZOP] should be undertaken to produce a comprehensive list of faults. This should cover: • Faults in all operational modes • Non-reactor faults [e.g., those associated with the Spent Fuel Pool (SFP) fuel route radioactive waste facilities] • Faults associated with essential support systems • Faults involving an initiating event and failure of one or more safety measures • Faults involving partial failures as well as total failures. • Faults arising from internal and external hazards.	For PCSR/PCER

FAP No.	Finding	Forward Actions	Delivery Phase
		The resulting list should be cross-checked with the faults considered in the PSA. New Design Basis Analysis (DBA) should be provided for newly identified faults that cannot be demonstrated to be bounded by existing analysis. This FAP item is connected with FAP item PSR 3-1 on numerical target development.	
PSR15.5-31	The offsite atmospheric dispersion factors (χ/Q) employed in the deterministic safety analysis [mainly sections 15.5.8 and 15.5.9] are established for a site outside of Great Britain (GB). The factors may not be broadly consistent with the conditions that could reasonably be expected for a GB Nuclear Power Plant (NPP) site. The relevant Safety Assessment Principle (SAP) is: ST.3.	Assumptions used in dose assessment calculations that will be broadly consistent with those that could reasonably be expected for a NPP site in GB should be identified. These will include distance of the reactor and other buildings with radiological inventory from the site boundary; and expected weather conditions. These should be compared with the assumptions currently used in the Deterministic Safety Analysis (DSA). If they are not bounded by the existing assumptions, the radiological calculations should be repeated using the new Generic Site Envelope (GSE) assumptions.	For PCSR/PCER
PSR15.5-32	Numerical Target Development Various sets of acceptance criteria used throughout PSR Ch. 15.5, such as: • 10 CFR 50 Appendix A General Design Criteria • NUREG-0800 • 10 CFR 50.34 No radiological criteria are identified for Design Extension Conditions (DECs). Much of the Design Basis Analysis (DBA) employs decoupling criteria to demonstrate the physical barriers to fission product release are maintained and which therefore, modulo activity in the coolant, meet all radiological criteria.	Develop a radiological criterion suitable for defining the set of faults subject to Design Basis Analysis (DBA) and judging the effectiveness of the safety measures designated in the DBA. The criterion does not need to be identical to Numerical Target 4 of the SAPs but should be broadly comparable to it. This FAP item is connected with item PSR15.5-33 on radiological consequence calculation methods.	For PCSR/PCER

FAP No.	Finding	Forward Actions	Delivery Phase
	In the UK, there is no prescription of such criteria, and so it is for the safety case to identify and justify them. However, ONR will make use of its own numerical targets whilst making its regulatory judgements. The associated Safety Assessment Principles (SAPs) are: Numerical Target 4, FA.5, and FA.7.		
PSR15-4	The Severe Accident Management strategy is an area of development for the BWRX-300. The information currently available on the Severe Accident Management provisions and the Severe Accident Analysis is considered insufficient for Step 2 GDA. Additional information is due to be available after PSR submission but before completion of the regulators’ assessment period in Step 2.	Submit the revisions of the following documents as submitted to CNSC: • SAA Methodology Report (007N3122 Rev 2) • SAA Sequence Selection Report, (007N6885 Rev B) • BWRX-300 Level 2 Probabilistic Safety Assessment (006N7608 Rev D) Also submit a licensing report, directly derived from the following document previously provided to ONR for information: • BWRX-300 Full Power Internal Event Severe Accident Analysis (DBR-0078529 Rev A)	January 2025 GDA Step 2
		Provide progress updates during Step 2 engagement as the next revisions of these documents are developed.	June 2025

APPENDIX C PRACTICAL ELIMINATION CLAIMS AND PROVISIONS

Practical elimination is achieved when the possibility of specific failures or plant conditions leading to an early or large radioactive release are shown to be extremely unlikely to arise with a high degree of confidence.

A practical elimination claim is made only for failures or conditions that cannot be mitigated by reasonably practicable means.

As a result of the adequate implementation of DL1, DL2, DL3, DL4a and DL4b features and functions, the likelihood is extremely low of an early or large off-site radioactive release. However, these PIEs and event sequences are mitigated by reasonably practicable means (the application of D-in-D); therefore, a specific practical elimination claim is not made relative to these PIEs and event sequences.

The aim of the practical elimination concept is to complement the implementation of D-in-D. Focused analysis is used to identify specific failures or plant conditions that cannot be practicably mitigated by application of defence-in-depth and could lead to unacceptable radiological consequences (a sudden rupture of the RPV is an example of such a failure). When such instances are identified, a specific practical elimination claim is required to substantiate that they are extremely unlikely to arise, with a high degree of confidence.

The practical elimination claims and provisions to achieve practical elimination are provided in Table C-1, Practical Elimination Claims and Provisions. See Appendix D for discussion on complementary design features.

Table C-1: Practically Eliminated Claims and Provisions

Practically Eliminated Claim	Provisions to Achieve Practical Elimination
The possibility that a sudden mechanical failure of RPV, in which the failure eliminates the capability of holding and cooling the core, is practically eliminated.	DL1 design and supporting operating provisions for a Reactor Coolant Pressure Boundary (RCPB) for all operating states within the plant design envelope, including DECs, of the highest reliability and quality according to American Society of mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC) throughout the lifetime provides greater confidence that the likelihood of sudden RPV failure is extremely unlikely.
The possibility of a single control rod falling out of the core is practically eliminated with several features that make it extremely unlikely.	DL1 design provision that the control blades include a bayonet style coupling design provision that requires a 45-degree rotation to uncouple, thus making it extremely unlikely for the control rod blade to become uncoupled from the drive during reactor operation. DL2 dual separation detection devices that sense if the hollow piston is no longer on the ball nut block control rod withdrawal and do not allow a separation distance to occur between the Fine Motion Control Rod Drive (FMCRD) and the control rod, or from the ball nut to an unlatched hollow piston. This essentially limits possible separation such that it is not physically possible for a control rod drop accident involving a single control rod falling completely out of the core to occur.
The possibility of direct containment heating following a postulated vessel failure is practically eliminated by reducing the likelihood of RPV failure at high pressure and by providing a strong containment design with complementary design features and design and supporting operating provisions.	DL4a ICS actuation provides greater confidence that RPV failure occurring at high pressure is extremely unlikely. DL4b RPV ultimate pressure regulation complementary design feature that provides a diverse means to limit the increase in reactor vessel pressure such that there are no credible core damaging accident sequences involving the core without cooling while the reactor is also pressurized, making direct containment heating following a postulated vessel failure highly unlikely with a high degree of confidence. DL4b complementary design feature of a containment ultimate overpressure vent ensures containment does not over-pressurise when RPV ultimate pressure regulation is actuated. DL1 design and supporting operating provisions for a strong containment for all states within the plant design envelope are of the highest reliability and quality according to ASME Boiler and Pressure Vessel Code throughout the lifetime.
The possibility of containment challenges from large steam explosions is practically eliminated by complementary design features and supporting operating provisions.	DL1 design provisions of a strong containment structure, components, and sufficiently large containment volume make containment load challenges from large steam explosions extremely unlikely. DL1 design and operating provisions ensure that inherent physical limitations associated with explosive interactions

Practically Eliminated Claim	Provisions to Achieve Practical Elimination
	(e.g., no subcooled water pool) are established and maintained.
The possibility for containment challenges because of combustible gas detonation is practically eliminated by complementary design features and supporting operating provisions.	DL1 design provision for an inert containment atmosphere and supporting combustible gas management operating provisions limiting concentrations well below detonation levels make containment challenges from combustible gas detonation extremely unlikely.
The possibility of containment failure from molten core concrete interaction is practically eliminated by design and supporting operating provisions.	DL1 design provision for a corium shield/liner supported by operating provisions confine and prevent the spread of molten core, making containment failure from molten core concrete interaction extremely unlikely.
The conditional containment failure by quasi-static over-pressurisation from a long-term loss of containment heat removal is practically eliminated by complementary design feature to vent containment.	DL4b complementary design feature of a containment vent and operating provisions make containment failure by quasi-static over-pressurisation from a long-term loss of containment heat removal extremely unlikely.
The possibility of an Interfacing System LOCA (ISLOCA) outside containment is practically eliminated by design provisions.	DL1 design provisions ensuring subsystems connected to the RCS are designed to an ultimate rupture strength at least equal to the RCS design pressure, making an ISLOCA very unlikely.
The condition of containment bypass consequential to Severe Accident (SA) progression is practically eliminated by design and operating provisions.	DL1 design and operating provisions for a RCPB for all states within the plant design envelope are the highest reliability and quality according to ASME Boiler and Pressure Vessel Code throughout the lifetime provides greater confidence that the likelihood of mechanical failure of the RPV isolation valve internals due to severe accident conditions is extremely unlikely.
The possibility of an SA with an open containment is practically eliminated by design and supporting operating provisions.	DL1 design provisions of a large inventory of available water and operating provisions and ample time for several diverse means to provide makeup water and prevent fuel from being uncovered in the fuel pool or a shutdown of the reactor during a shutdown condition with the containment open provides greater confidence that the likelihood of an SA with an open containment is extremely unlikely.
The possibility a high pressure melt ejection with potential for debris dispersion such that debris is not retained in the lower containment is practically eliminated by complementary design features and design and supporting operating provisions.	DL4b ultimate pressure regulation complementary design feature that provides a diverse means to limit the increase in reactor vessel pressure ensures that there are no credible core damaging accident sequences involving the core without cooling while the reactor is also at high pressure, making containment failure from high pressure melt ejection extremely unlikely. DL4b complementary design feature of a containment vent and operating provisions make containment failure by quasi-

Practically Eliminated Claim	Provisions to Achieve Practical Elimination
	static over-pressurisation from a high pressure melt ejection extremely unlikely. DL1 design provision for a corium shield/liner supported by operating provisions confine and prevent the spread of a molten core, making containment failure from molten core concrete interaction extremely unlikely.
Any credible possibility of an out of core criticality event with fuel handling is practically eliminated by design provisions and supporting operating provisions.	DL1 design provisions and supporting operating provisions provide design and administrative controls that effectively manage factors that influence system reactivity and the likelihood of criticality. These factors are enrichment, mass, moderation, geometry, reflection, interaction, and spacing. Plant procedures prohibit the handling and storage of more fuel assemblies than have been determined to be safely subcritical under the most adverse moderation conditions feasible in unborated water. The amount of water moderator available for internal flooding is limited by the design of the storage location structures thus making an out of core criticality event extremely unlikely.

APPENDIX D COMPLEMENTARY DEFENCE LINE 4 FUNCTIONS FOR MITIGATING DESIGN EXTENSION CONDITIONS

Complementary design features are used to prevent accident progression or mitigate the consequences of DECs. Complementary design features mitigating functions are provided in Table D-1, Complementary Design Features.

Complementary design features are finalized as the Beyond Design Basis Analysis (BDBA) PSA Level 2 analysis progresses.

Table D-1: Complementary Design Features

DEC Function	Complementary Design Feature
Reactivity Control	DL4a complementary design feature diverse protection system actuation logic diverse and independent from the DL3 Reactor Protection Systems actuation logic provides greater confidence that the condition is extremely unlikely to progress to a severe accident condition. DL4a complementary design feature of the FMCRD motor run-in function provides greater confidence that the condition is extremely unlikely to progress to a severe accident condition. DL4a complementary design feature of alternate control rod insertion to ensure that Hydraulic Control Unit (HCU) scram pressure is released to the FMCRD even in the unlikely event of failure of the scram solenoid pilot valves, provides greater confidence that the condition is extremely unlikely to progress to a severe accident condition. DL4b ultimate pressure regulation complementary design feature that provides a diverse means to limit the increase in reactor vessel pressure for the extremely unlikely condition, thus making failure propagation from the RPV to the containment extremely unlikely. DL4b Boron Injection System (BIS) complementary design feature provides a means to place the plant in a stable, controlled configuration if control rods fail to insert. DL1 design provision that the control blades include a bayonet style coupling that requires a 45-degree rotation to uncouple, makes it extremely unlikely for the control rod blade to become uncoupled from the drive during reactor operation. DL2 dual separation detection devices that sense if the hollow piston is no longer on the ball nut block control rod withdrawal and do not allow a separation distance to occur between the FMCRD and the control rod or from the ball nut to an unlatched hollow piston. This essentially limits possible separation such that it is not physically possible for a control rod drop accident involving a single control rod falling completely out of the core to occur. DL1 design provisions and supporting operating provisions provide design and administrative controls that effectively manage factors that influence system reactivity and the likelihood of criticality. These factors are enrichment, mass, moderation, geometry, reflection, interaction and spacing. Plant procedures prohibit the handling and storage of more fuel assemblies than have been determined to be safely subcritical under the most adverse moderation conditions feasible in unborated water. The amount of water moderator available for internal flooding is limited by the design of the storage location structures thus making any credible possibility of an out of core criticality extremely unlikely.
RPV Depressurization	DL4b ultimate pressure regulation complementary design feature that provides a diverse means to limit the increase in reactor vessel pressure ensure that there are no credible core damaging accident sequences involving the core without

DEC Function	Complementary Design Feature
	cooling while the reactor is also pressurized, making direct containment heating following a postulated vessel failure or high pressure melt ejection extremely unlikely.
In-Vessel Core Cooling	None required because in-vessel retention is not considered for severe accidents.
Cooling of Corium Debris in the Containment Corium Shield/Liner	DL1 design provision for a corium shield/liner supported by operating provisions confine and prevent the spread of a molten core, making containment failure because of molten core concrete interaction extremely unlikely. The corium/shield liner contains the melt debris and provides a means of cooling the debris bed to prevent core concrete interaction and ultimate containment over-pressurisation. DL4b design provisions for corium shield cooling are provided. This is provided by a primary containment system piping connection routing water from the reactor cavity to the corium shield.
Cooling of High Pressure Melt Ejection Debris	Dispersal of debris material outside the corium shield/liner is limited by eliminating energetic phenomena like high pressure melt ejection so that it does not contribute to a significantly elevated containment temperature transient.
Containment Isolation	DL4a containment isolation actuation function provides greater confidence that containment isolation failure in the event of a CCF of DL3 isolation functions is extremely unlikely.
Containment Pressure Control: Heat Removal	DL1 design provision for passive containment cooling provides a means to remove decay heat.
Containment Pressure Control: Venting	DL4b complementary design feature of a containment vent and operating provisions make containment failure by quasi-static over-pressurisation from a long-term loss of containment heat removal is extremely unlikely.
Combustible Gas Control	DL1 design provision for an inert containment atmosphere and supporting combustible gas management operating provisions limiting concentrations well below detonation levels make containment challenges because of combustible gas detonation is extremely unlikely.
Post-Accident Monitoring	DL1 design provision for post-accident monitoring provides information to facilitate post-accident response and evaluation of RPV and containment conditions.

APPENDIX E APPROACH TO DEVELOPMENT OF THE FAULT SCHEDULE

This section concerns the proposals for development of a UK type fault schedule. FAP item PSR 15.5-28 from the FAP schedule (Reference 15.9-13), as described in Appendix B, directly pertains to this Appendix.

Our Understanding of Expectations for Fault Schedules

Long standing RGP in the UK is the summarising of key aspects of a safety case, in particular the DBA, in a table commonly referred to as a *fault schedule*. ONR's SAPs (Reference 15.9-11) ESS.11 and FA.8 encode the expectation for a fault schedule.

The principal objective of a fault schedule is to provide clear auditable linkage, sometimes referred as the golden thread, between faults, fault sequences, and safety measures, and must, present the following information:

- Initiating Event
- Initiating Event Frequency.
- Unmitigated and unprotected consequences.
- Fundamental safety functions to be delivered [including control of reactivity, cooling, and containment].
- Safety measures Structures, Systems, and Components (SSCs) and human actions] to deliver the safety functions.

It is the fault schedule that yokes together the discrete DBA into a coherent safety case robustly demonstrating the fault tolerance of the engineering and effectiveness of the safety measures. In this way, it is a powerful way to demonstrate the completeness of the list of faults.

Beyond these fundamental requirements, the fault schedule is readily extended to incorporate additional useful, pertinent information, such as:

- The operating mode and plant configuration assumed for the initiating event
- Categorisation of associated safety functions and classification of safety measures
- Decomposition of safety functions and safety measures
- Minimum number of trains required to deliver the safety functions
- Number of trains available in specific operating modes/plant configuration
- D-in-D measures additional to the claimed design basis measures
- Essential support systems [e.g., power, cooling water] required by the design basis measures
- The type of each safety measure [passive, automatic, manually initiated]
- The key parameters, including operating rules, and control and instrumentation system that initiates operation of a safety measure
- Specification of all faults bounded by the events included in the fault schedule
- Links to hazard and fault identification, such as Failure Modes and Effects Analysis, or Hazard and Operability Studies
- Links to initiating event frequency substantiation
- Links to supporting transient analysis and narrative

- Links to additional engineering details and substantiation

This additional information can efficiently show that all operating modes have been considered, and safety functions are appropriately categorised and safety measures appropriately classified, as well detailing the minimum availability requirements for systems with inbuilt redundancy. It also greatly improves the navigability of the safety case.

Although the principal aim of fault schedule is to summarise the DBA, extending it to include faults and fault sequences outside the design basis region can be a powerful way to demonstrate the completeness of the safety case and existence of D-in-D within the facility.

The fault schedule provides a link from the requirements identified in the fault analysis to the engineering which delivers those requirements.

In addition to its summarisation role, for a facility under design development, the fault schedule also performs a configuration control role, and provides a means of communication between the safety case and engineering design teams.

Current Position

As part of its design development process, GEH makes use of a Fault List. This presents a single, consolidated route map to all scenarios composing the complete set of DSAS for the BWRX-300. Throughout the design development process, it is used to support organized iteration between design and analysis teams. It is updated throughout this process to reflect the current state of design and analysis maturation. In its final state, it will support the efficient validation that all required DL functions have been identified and provide traceability between the DL functions and those analysis cases that establish their performance bases.

The extant Fault List comprises the following fields:

- **Fault Sequence Identifier.** This uniquely identifies the fault sequence and encodes the fault group, the PIE, additional conditions, and fault sequence category. For example, “PI-LR-TT_CCF-DL2_CN-DBA” encodes the generator (LR-TT, which is in the PI group, which assumes a Common Cause Failure (CCF) of the DL2 technology platform (CCF-DL2), and which is analysed using conservative (CN) DSA as a design basis accident (CN-DBA).
- **Initial Plant Mode indicates** the operating states of the plant that were applicable at the time the PIE occurs
- **Reactor Mode Switch Position** indicates the reactor mode switch position at the time the PIE occurs
- **Fault Sequence Name** This is a descriptive title of the fault sequence
- **Postulated Initiating Event** This is the initiating event of the fault sequence
- **Additional Conditions or Failures** This includes failures in mitigation functions, along with additional assumed conditions defining the fault sequence
- **Fault Sequence Category** This is category of the fault sequence being analysed, for example AOO, DBA, or DEC
- **Fault Sequence Summary** is narrative summary of the expected sequence of events, starting with the PIE and finishing with the achievement of a controlled state
- **Deterministic Safety Analysis** is the type of DSA used, for example: Baseline, CN, or Extended
- **Control of Reactivity – DL** This identifies the DL function credited in delivering the Control of Reactivity Fundamental Safety Function (FSF), for example DL3-04

- **Control of Reactivity – FSF** This presents the name of the safety function corresponding to the previous column, for example Hydraulic Scram on High Neutron Flux
- **Fuel Cooling – DL** This identifies the DL function credited in delivering the Fuel Cooling FSF for example DL2-09
- **Fuel Cooling – FSF** This presents the name of the safety function corresponding to the previous column, for example Turbine Bypass Valve Fast Open on Turbine Trip Demand
- **Long Term Heat Removal – DL** This identifies the DL function credited in delivering the Long-Term Heat Removal FSF, for example DL4a-30
- **Long Term Heat Removal – FSF** This presents the name of the safety function corresponding to the previous column, for example Cavity Pool Makeup from ICS on RPV Water Level Reduction (L9)
- **Confinement of Radioactive Materials – DL** This identifies the DL function credited in delivering the Confinement of Radioactive Materials FSF, for example DL3-20
- **Confinement of Radioactive Materials – FSF** This presents the name of the safety function corresponding to the previous column, for example Main Steam RIV/MSRIVC Isolation on Main Steam Line Break Indication
- **DL 1 Requirements and Development Assumptions.** This column records DL1 design assumptions which underpin the validity of the specific PIE or fault sequence being evaluated and includes general and functional requirements
- **Comments** This records any additional information related to the fault sequence, such as: design assumptions, areas of uncertainty, or questions to be answered with future work

Proposals for Developing a Fault Schedule

It is proposed to develop a fault schedule in two phases. The first phase will be to supplement the existing fault list and the second phase will be the development of a bespoke fault schedule. Because the fault list and fault schedule are summarizations of other parts of the safety case, principally the deterministic safety analysis, their development will be linked to the development of the safety case.

Phase 1 – Extension of the Fault List

Additional columns will be added to the extant fault list to provide the following information:

- Initiating event frequencies
- The safety measures (SSCs and human actions) designated to mitigate each event

Within the existing fault list fields, additional detail will be added to indicate whether a fault is bounded or bounding, along with signposting to the bounding or bounded faults respectively.

This work will proceed in parallel with the natural ongoing design and safety case development work, in particular work to complete the set of faults considered in the safety case.

Because of the manner in which internal and external hazards are currently considered in the safety case, they will not be included in this phase of the work.

Phase 2 – Development of a Bespoke Fault Schedule

The result/work/information of phase one will be built on to develop as the following additional information will be incorporated:

- Categorisation of associated safety functions and classification of safety measures
- Minimum number of trains required to deliver the safety functions
- Number of trains available in specific operating modes / plant configuration
- The type of each safety measure [passive, automatic, manually initiated]
- D-in-D measures additional to the claimed design basis measures
- Essential support services

Links to the material in the rest of the safety case, such as fault identification; initiating event frequency derivation; transient analysis; and engineering substantiation will be developed.

The details of the presentation are still to be developed, but GEH considers the fault schedule developed by GEH for the UK ABWR to be an exemplar. The use of bold, italic, and coloured fonts will be supplemented to improve accessibility. An example fault schedule header for the BWRX-300, is shown overleaf in Table E-1.

Documentation will be developed, as part of the safety case manual, to provide guidance on development of the fault schedule, its relation to other activities such as: fault sequence modelling, grouping of fault sequences, consequence derivation, categorisation and classification, engineering substantiation, and the development of operating rules.

Table E-1: Exemplar Fault Schedule Headings

Fault ID	Initial Plant Mode(s)	Reactor Mode Switch Position	Bounding Fault Name	Postulated Initiating Event (Bounding Fault)	Other Faults	Fault Sequence Category	Fault Sequence Summary (including consequential loss)	Deterministic Safety Analysis	Control of Reactivity		Fuel Cooling		Long Term Heat Removal		Confinement of Radioactive Materials		DL1 Requirements and DL Functional Assumptions	Comments
									DL	FSF	DL	FSF	DL	FSF	DL	FSF		
1.																		
1.1																		