GE Hitachi Nuclear Energy

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BWRX-300 UK Generic Design Assessment (GDA) Chapter 15.1 – Safety Analysis – General Considerations

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EXECUTIVE SUMMARY

This document is Chapter 15.1, General Considerations, of the Preliminary Safety Report of the GEH BWRX-300 for the purposes of UK Generic Design Assessment. It presents the introduction and general considerations of the safety analyses which appears in Chapter 15.

ACRONYMS AND ABBREVIATIONS

Acronym	Explanation
ALARP	As Low As Reasonably Practicable
AOO	Anticipated Operational Occurrence
BDBA	Beyond Design Basis Accident
BL-DSA	Baseline Deterministic Safety Analysis
CCF	Common Cause Failure
CN-DSA	Conservative Deterministic Safety Analysis
DBA	Design Basis Accident
DEC	Design Extension Condition
D-in-D	Defence-in-Depth
DL	Defence Line
DSA	Deterministic Safety Analysis
EHE	External Hazard Evaluation
EX-DSA	Extended Deterministic Safety Analysis
FAP	Forward Action Plan
FFA	Functional Failure Analysis
FMEA	Failure Mode and Effects Analysis
FSF	Fundamental Safety Function
GDA	Generic Design Assessment
GEH	GE Hitachi Nuclear Energy
HFEA	Human Failure Event Analysis
IAEA	International Atomic Energy Agency
IH	Internal Hazard
IHE	Internal Hazard Evaluation
NPP	Nuclear Power Plant
ONR	Office of Nuclear Regulation
PIE	Postulated Initiating Event
PSA	Probabilistic Safety Assessment
PSR	Preliminary Safety Report
RGP	Relevant Good Practice
RPV	Reactor Pressure Vessel
SA	Severe Accident
SAA	Severe Accident Analysis
SSC	Structures, Systems, and Components

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REVISION SUMMARY

Revision #	Section Modified	Revision Summary
A	All	Initial Revision

15.1 GENERAL CONSIDERATIONS

Chapter Route Map

This subchapter is part of PSR Ch. 15 which presents the BWRX-300 Safety Analysis and comprises the following subchapters:

- PSR Ch. 15.1 – Safety Analysis - General Considerations *[this subchapter]*
- PSR Ch. 15.2 – NEDC-34180P, "BWRX-300 UK GDA Ch. 15.2: Safety Analysis - Identification, Categorisation, and Grouping of Postulated Initiating Events (PIEs) and Accident Scenarios," (Reference 15.1-1)
- PSR Ch. 15.3 – NEDC-34181P, "BWRX-300 UK GDA Ch. 15.3: Safety Analysis Safety Objectives and Acceptance Criteria," (Reference 15.1-2)
- PSR Ch. 15.4 – NEDC-34182P, "BWRX-300 UK GDA Ch. 15.4: Safety Analysis Human Actions," (Reference 15.1-3)
- PSR Ch. 15.5 – NEDC-34183P, "BWRX-300 UK GDA Ch. 15.5: Deterministic Safety Analysis," (Reference 15.1-4)
- PSR Ch. 15.6 – NEDC-34184P, "BWRX-300 UK GDA Ch 15.6: Probabilistic Safety Assessment," (Reference 15.1-5)
- PSR Ch. 15.7 – NEDC-34185P, "BWRX-300 UK GDA Ch. 15.7: Deterministic Safety Analyses Analysis of Internal Hazards," (Reference 15.1-6)
- PSR Ch. 15.8 – NEDC-34186P "BWRX-300 UK GDA Ch. 15.8: Safety Analysis External Hazards," (Reference 15.1-7)
- PSR Ch. 15.9 – NEDC-34187P, "BWRX-300 UK GDA Ch. 15.9: Summary of Results of the Safety Analyses," (Reference 15.1-8)

This layout mainly follows the structure set out in International Atomic Energy Agency (IAEA) SSG-61 "Format and Content of the Safety Analysis Report for Nuclear Power Plants, Specific Safety Guide Series," (Reference 15.1-9) with the exception that Internal and External Hazards are discussed in two separate subchapters.

A synopsis of the contents of these subchapters is presented below.

Subchapter Structure

This subchapter presents the general description for the BWRX-300 Safety Analyses and comprises the following main sections:

- 15.1 – Introduction *[this section]*
- 15.1.1 – Scope of Safety Anlysis and Approach Adopted
- 15.1.2 – Analysis of Design Basis Conditions
- 15.1.3 – Analysis of Design Extension Conditions
- 15.1.4 – Design Extension Conditions without Core Damage
- 15.1.5 – Design Extension Conditions with Core Damage
- 15.1.6 – Analysis of Hazard
- 15.1.7 – Practical Elimination
- 15.1.8 – Explanation of the Structure of Chapter 15

Interfaces with other Chapters

This subchapter interfaces directly with the following Preliminary Safety Report (PSR) Chapters:

- PSR Ch. 15 – Safety Analysis (all other subchapters)

Purpose

The purpose of this subchapter is to provide an introduction to the safety analyses of the BWRX-300 which are presented in PSR Ch. 15 as a whole, covering both Deterministic Safety Analysis (DSA) and Probabilistic Safety Assessment (PSA), human factors considerations, and internal and external hazards. It includes a description of the scope of the safety analysis and the approach adopted for each plant state, from normal operation to design extension conditions with core melting.

Scope

The scope of this subchapter comprises the safety analyses presented in NEDC-34183P (Reference 15.1-4) and NEDC-34184P (Reference 15.1-5), and the evaluation of hazards in NEDC-34185P (Reference 15.1-6) and NEDC-34186P (Reference 15.1-7).

Country Specific Material – UK Step 2 GDA

The PSR is being submitted as part of Step 2 of the Generic Design Assessment (GDA) by the United Kingdom (UK) Office for Nuclear Regulation (ONR). The GDA is a generic, non-site-specific assessment of a standard BWRX-300 Nuclear Power Plant (NPP) design. It is intended to determine whether a proposed reactor type could be constructed, operated, and decommissioned in the UK. Step 2 is a fundamental assessment of the generic safety, security, and environment protection cases. It is intended to judge these aspects to identify potential issues 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 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 SubPSR Ch. 15.1 does not directly support any forward actions, as per Appendix B Forward Action Plan. PSR Ch. 15 subchapters that do identify Forward Action Plan (FAP) items are presented in FAP schedules in their respective subchapters. Subchapters with FAP items are: NEDC-34182P (Reference 15.1-3), NEDC-34183P (Reference 15.1-4), NEDC-34184P (Reference 15.1-5), NEDC-34185P (Reference 15.1-6) and NEDC-34186P (Reference 15.1-7). 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.

15.1.1 Scope of Safety Analysis and Approach Adopted

High Level Strategy

The BWRX-300 is founded on the principle of Defence-in-Depth (D-in-D) which is consistent with IAEA guidance, SSR-2/1, "Safety of Nuclear Power Plants: Design, Specific Safety Requirements," (Reference 15.1-10) and used to formulate the safety strategy, NEDC-33934P, "BWRX-300 Safety Strategy," (Reference 15.1-11). D-in-D is the provision of multiple layers of defence against exposure of workers and of the public to radioactivity exceeding levels determined to be safe.

In the BWRX-300 safety strategy, there are two aspects to this. Firstly, there are the physical barriers in place to prevent the release of radioactivity. The integrity of one or more of these physical barriers needs to be maintained to prevent unacceptable releases. The fuel cladding, reactor coolant pressure boundary, and containment are examples of these physical barriers.

The second aspect is the combination of active, passive, and inherent safety features and functions, as well as design and operational practices, which minimise challenges to the physical barriers, maintain integrity of the barriers when challenged, and were a barrier to be breached, ensure the integrity of the remaining barriers.

The features, functions, and practices that protect the integrity of the barriers are termed Defence Lines (DLs). The objectives of the DLs are set out in Table 15.1-1:

The relation between the two aspects is illustrated in Figure 15.1-1:

A multi-faceted, integrated approach has been used, making use of a robust fault evaluation process and multiple analyses to provide bases for a comprehensive set of design to analysis requirements, which inform design development and modification. These evaluations and analyses are performed iteratively as the design and safety demonstration activities are developed.

The framework is illustrated in Figure 15.1-2, and fully explained in NEDC-33934P (Reference 15.1-11).

General Approach

A critical part of the safety analysis is to systematically and comprehensively identify Structures, Systems, and Components (SSCs) functional and human failures that initiate a PIE, initiate a hazard that leads to a PIE or worsen a hazard that leads to a PIE, NEDC-33934P (Reference 15.1-11) process identifies two plant-level failure analyses to identify such failures:

- Functional Failure Analysis (FFA)—The FFA identifies failures of plant systems or equipment with potential to cause a challenge to an FSF. Potential failures are identified in Failure Mode and Effects Analyses (FMEA) performed on each plant system. The FFA is limited to random single failures and CCFs.
- Human Failure Event Analysis (HFEA)—The HFEA identifies failures that involve a single human failure event that could potentially lead to a PIE, specifically those that could initiate an abnormal or accident event sequence. Human failure events involve erroneous decisions or human action(s) that lead to an unplanned plant transient and typically involve unplanned changes to plant equipment status by equipment operators or maintenance personnel.

The scope of each failure analysis encompasses the complete range of normal plant states (i.e., full power, low power, shutdown, and refuelling) as the type and consequence of each failure may differ depending on the plant state. The analyses also consider all sources of radioactivity (e.g., spent fuel, fuel being handled, radioactive waste, activated material) in addition to the reactor core itself. The output of the plant-level failure analyses (i.e., potential PIE initiators) is passed on to the fault evaluation process, where PIEs are systematically identified, categorised, and grouped together using different types of PIE and event sequence selection processes, which include both deterministic and probabilistic inputs, as described in NEDC-34180P (Reference 15.1-1).

In addition to failure analyses, hazard evaluations are performed to ensure that all potential hazards the plant might experience are identified and considered, NEDC-34180P (Reference 15.1-11) identifies two hazard evaluations:

- Internal Hazard Evaluation (IHE)
- External Hazard Evaluation (EHE)

A primary objective of each hazard evaluation is to identify hazards with the potential to initiate a PIE and pass that list of hazards to the relevant downstream analysis, PSA and/or the deterministic hazards analyses (e.g., seismic hazard analyses, fire hazards assessment, pipe rupture hazards analysis). The focus of the EHE and IHE is identifying the list of credible hazards and defining the expected frequencies of those credible hazards. The plant is designed to withstand the hazards while maintaining performance of the FSFs through implementation of DL1 design requirements.

Following PIE and event sequence selection, DSA is performed. The DSA objectives include:

- Demonstrating the design meets the acceptance criteria established following a graded approach for each plant state. The graded approach application may lead to acceptance criteria more restrictive for events with higher occurrence probability.
- Provide analytical basis to support the derivation of the plant technical specifications for normal operation
- Provide analytical basis for establishing and validating accident management procedures and guidelines

The safety analyses scope plant state includes the following plant states:

- Normal operation
- Anticipated Operational Occurrences (AOOs)
- Design Basis Accidents (DBAs)
- Design Extension Conditions (DECs) with or without core damage (i.e., Beyond Design Basis Accidents (BDBAs))

The BWRX-300 DSA uses a layered analysis approach that includes three types of DSA evaluations:

- Baseline Deterministic Safety Analysis (BL-DSA)
- Conservative Deterministic Safety Analsysi (CN-DSA)
- Extended Deterministic Safety Analysis (EX-DSA)

These DSA acceptance criteria are discussed in NEDC-34181P (Reference 15.1-2). The DSA results are compared against the applicable plant state acceptance criteria and dose limits.

The PSA is performed to complement the DSA. PSA estimates the overall risk presented by the facility that is compared to the safety goals specified in NEDC-34181P (Reference 15.1-2). The PSA is presented in NEDC-34184P (Reference 15.1-5).

15.1.2 Analysis of Design Basis Conditions

The BWRX-300 design basis conditions are normal operations, AOOs and DBAs described below:

- Normal Operation is operation within specified operational limits and conditions (see NEDC-34188P, "BWRX-300 UK GDA Ch. Ch. 16: Operational Limits and Conditions of Safe Operation," (Reference 15.1-12) and includes the full range of plant operating modes.
- AOOs are deviations from normal operation that are expected to occur at least once during the operating lifetime of the reactor facility. The objective of the AOO safety analysis (i.e., BL-DSA) is to demonstrate that DL2 functions are effective for most AOO PIEs in meeting the applicable acceptance criteria.
- DBAs conditions are identified as deviations from normal operations that are less frequent and more severe than AOOs. An objective of DBA safety analysis (i.e., CN-DSA) is to demonstrate that DL3 functions are effective in mitigating events and meeting the applicable acceptance criteria.

Acceptance criteria applicable to the DSA for each plant state is discussed in NEDC-34181P (Reference 15.1-2). The response to AOOs and DBAs is achieved by SSCs specifically designed to mitigate these events by performing their DL2 and DL3 functions (see NEDC-34165P, "BWRX-300 UK GDA Ch. 3: Safety Objectives and Design Rules for SSCs," (Reference 15.1-13).

15.1.3 Analysis of Design Extension Conditions

DECs are postulated accident conditions that are less frequent than DBAs. DECs may occur with or without core damage.

15.1.4 Design Extension Conditions without Core Damage

EX-DSA is performed for DECs without core damage demonstrating that releases of radioactive material are kept within acceptable limits and support the PSA determination of no core damage.

The EX-DSA assesses plant performance in response to PIEs and event sequences such as:

- PIEs caused by single equipment failures that are in the DEC event category (i.e., low frequency events)
- PIEs initiated by a Common Cause Failure (CCF) of DL3
- Multiple failures defined as complex sequences identified in the Level 1 PSA
- When AOO PIEs are mitigated using DL2 and DL3 hydraulic scram functions in the BL-DSA and CN-DSA, a DEC event sequence is analyzed, assuming the same AOO PIE plus an assumed CCF of the mechanical equipment providing motive force for hydraulic scram.
- DBA PIEs caused by a single equipment failure, for which DL3 functions were credited in both the BL-DSA and CN-DSA. The same DBA PIE is analyzed assuming DL3 CCF.

The results of the EX-DSA for DECs without core damage are discussed in NEDC-34187P (Reference 15.1-8).

15.1.5 Design Extension Conditions with Core Damage

DECs with core damage are referred to as Severe Accidents (SA) and involve a catastrophic failure, core damage, and fission product release. A SA is generally considered to begin with the onset of core damage. To the extent that core damage is not practically eliminated, representative DECs with core damage are postulated to provide inputs for the containment design and safety features ensuring containment functionality. This set of accidents is considered in the design of corresponding safety features for DECs and represents a set of bounding cases. Accident scenarios considered for practical elimination are described in NEDC-34187P, Appendix 15A (Reference 15.1-8).

SA sequences are selected that identify representative core damage scenarios and corresponding plant damage states that are used as the basis for performing the Severe Accident Analysis (SAA). The scope of SA sequence selection corresponds to sequences involving significant core damage that could lead to a containment breach and radioactive release analysed in the Level 2 in NEDC-34184P (Reference 15.1-5). The selected SA sequences are included in the fault list.

The SAA goal is to provide input to accident management for terminating the progression of core damage, maintaining containment integrity as long as possible, and minimizing on-site and offsite radioactive material releases. Halting core damage progress prevents Reactor Pressure Vessel (RPV) failure.

Any system or equipment that remains capable of performing its Safety Category function, given the context of the event sequence, can be credited during SAA.

15.1.6 Analysis of Hazards

15.1.6.1 Analysis of Internal Hazards

Internal Hazards (IHs) are hazards that arise from within the site boundary, and that results in the failure of operations or facilities that are under the control of the operating organisation. The IHE identifies conditions originating within the boundaries of the site, and with potential to lead to an unplanned plant transient. The IH condition does not directly challenge a Fundamental Safety Function (FSF) (like in FFA), but the effects of the hazard may cause equipment failures.

The IHE addresses hazards from individuals, and a combination of sources, and organises them by quantitative frequency. The primary objective of the IHE is to identify IHs with the potential to initiate a PIE, and pass the list of credible IHs to the deterministic hazard analyses and PSA. This is to demonstrate that the plant design can withstand the hazards while maintaining performance of the FSFs.

Deterministic hazards analyses are described in other PSR sections as required. These IHs analyses include:

- Fires (discussed in NEDC-34171P, “BWRX-300 UK GDA Ch. 9A: Auxiliary Systems,” (Reference 15.1-14))
- Explosions, missiles from rotating or pressurised equipment (discussed in NEDC-34165P (Reference 15.1-13))
- Collapse of structures/falling objects (discussed in NEDC-34165P (Reference 15.1-13)
- Pipe whip, jet effects, and flooding (discussed in NEDC-34165P (Reference 15.1-13))

15.1.6.2 Analysis of External Hazards

External hazards are natural, and human-induced hazards that originate from a source that is not under control of the NPP license holder. The EHE addresses individual hazard sources and combinations of sources and organises them by quantitative frequency. The design basis hazard frequencies and magnitudes are selected considering the regional location of the facility and regulatory expectations. Types of external hazards include:

- Natural external hazards include earthquakes, droughts, floods, high winds, tornadoes, tsunami, and extreme meteorological conditions

- Human-induced external hazards include toxic gas releases, aircraft crashes, or ship collisions

The primary objective of the EHE is to identify external hazards with the potential to initiate a PIE, define the expected frequency, and pass the list of credible external hazards to the deterministic hazard analyses and PSA. Similarly to the IHE, this is to demonstrate that the plant design can withstand the external hazards while maintaining performance of the FSFs.

Once the external hazards are identified, the BWRX-300 structures are designed to withstand these external hazards, and the resulting protection against these hazards. External events are site-specific and are specified in the site evaluation provided in NEDC-34164P, “BWRX-300 Ch. 2: Site Characteristics,” (Reference 15.1-15).

15.1.7 Practical Elimination

Practical elimination is applied to events or sequences of events leading to or involving core damage, (a SA) where confinement of radioactive materials cannot be reasonably achieved. Event sequences that are either physically impossible or extremely unlikely to occur are considered for practical elimination.

The practical elimination demonstration is performed with accident conditions and phenomena knowledge and is substantiated by relevant evidence (see NEDC-34187P, Appendix A (Reference 15.1-8)).

15.1.8 Explanation of the Structure of Chapter 15

PSR Ch. 15 has been split into the following subchapters:

PSR Ch. 15.1 – General Considerations. This subchapter provides an introduction to the safety analysis chapter, covering both DSA and PSA. This subchapter includes a description of the scope of the safety analysis and the approach adopted for each plant state, for normal operations to design extension conditions with core melting. This subchapter also explains the approach to the fault analysis and hazards evaluation to identify hazards that could lead to PIEs.

PSR Ch. 15.2 – Identification, Categorisation, and Grouping of PIEs and Accident Scenarios, NEDC-34180P (Reference 15.1-1). This subchapter provides the approach used to identify PIEs and accident scenarios for the DSA. It also details the method of categorisation and grouping of PIEs as a fundamental element of fault evaluation and event sequence selection.

PSR Ch. 15.3 – Safety Objectives and Acceptance Criteria, NEDC-34181P (Reference 15.1-2). This subchapter provides the safety objectives of the PSR for the GDA of the BWRX-300. It presents the acceptance criteria (quantitative and qualitative) for DSA and PSA that will be used in NEDC-34183P (Reference 15.1-4) and NEDC-34184P (Reference 15.1-5).

PSR Ch. 15.4 – Human Actions, NEDC-34182P (Reference 15.1-3). The purpose of this subchapter is to describe the approach to identify and model human actions in the BWRX-300 DSA and PSA. It also describes the approach to substantiation of human actions. The subchapter will present a level of detail commensurate with a 2 step GDA (claims and arguments only) and will be structured in line with IAEA, SSG-61 (Reference 15.1-9) (noting that SSG-61 does not differentiate between the level of detail required in PSR and later more detailed safety reports). NEDC-34190P, “BWRX-300 UK GDA Ch. 18: Human Factors Engineering,” (Reference 15.1-16) also incorporates material related to Human Factors.

PSR Ch. 15.5 – Deterministic Safety Analysis, NEDC-34183P (Reference 15.1-4). This subchapter defines the initiating events that are reasonably foreseeable, conservatively justifies accident sequences that follow those PIEs and assesses the design against engineering principles. The purpose is to demonstrate the fault- tolerance of the design, the effectiveness of the safety measures, and to support the claim that all risks associated with the design and its operation have been reduced ALARP. The DSA does not quantify risk; instead, the adequacy of the design and the suitability and sufficiency of the safety measures are assessed against acceptance criteria.

PSR Ch. 15.6 – Probabilistic Safety Assessment, NEDC-34184P (Reference 15.1-5). This subchapter provides a description of the PSA that has been undertaken for the BWRX-300, with an overview of the results and comparison with the safety goals and numerical targets. This subchapter will present a discussion relating to the PSA has and will continue to support risk-informed design and decision making and support the claim that the BWRX-300 risk is ALARP.

PSR Ch. 15.7 – Internal Hazards, NEDC-34185P (Reference 15.1-6). This subchapter will provide a description of the IH to be considered within the BWRX-300 GDA PSR. This subchapter will explain the identification process, assessment methodologies, and demonstrate the tolerance for the IHs of the BWRX-300 design.

PSR Ch. 15.8 – External Hazards, NEDC-34186P (Reference 15.1-7). This subchapter will provide a description of the derivation of External Hazard (EH) to be considered within the BWRX-300 GDA PSR. It will explain the process used to systematically identify and screen natural and man-made hazards. It will summarise the measures inherent in the design to ensure that the FSFs and the SSCs that deliver them are protected against design basis external hazards and combinations thereof.

PSR Ch. 15.9 – Summary of Results, NEDC-34187P (Reference 15.1-8). This subchapter provides a summary of the overall results of the safety analysis for each of the categories of events and covers DSA and PSA. This subchapter should also confirm that all relevant nuclear plant design expectations have been met, and a route for completion of any outstanding aspects should be defined.

NEDC-34187P (Reference 15.1-8) contains three appendices to PSR Ch. 15:

Appendix A – Reference Source Term for Conditions that are Practically Eliminated. This appendix discusses the accident scenarios considered for practical elimination.

Appendix B – Risk Reduction Included as Defence Line 4 Functions for Mitigating Design Extension Conditions. This appendix presents the risk reduction features included in Defence Line 4 to mitigate DECs.

Appendix C – Approach to the Development of the Fault Schedule. This appendix presents the proposed approach to the development of the extant fault list into a fault schedule, along with details of how it will be used during design development.

Table 15.1-1: Overview of Defence Line Objectives

DL	Objective
DL1	Minimise potential for failures and initiating events to occur in the first place and minimise potential for failures to occur in subsequent lines of defence.
DL2	Actively control key plant parameters associated with FSFs and detect and mitigate AOOs.
DL3	Detect and mitigate DBA PIEs and failure of DL2 functions.
DL4a	Detect and mitigate DECs, including event sequences associated with some DBA PIEs and failure of DL3 functions.
DL4b	Detect and mitigate DECs to prevent core damage or mitigate the consequences of core damage events (SA).
DL5	Employ emergency preparedness measures to protect the public from consequences of significant releases of radioactive materials.

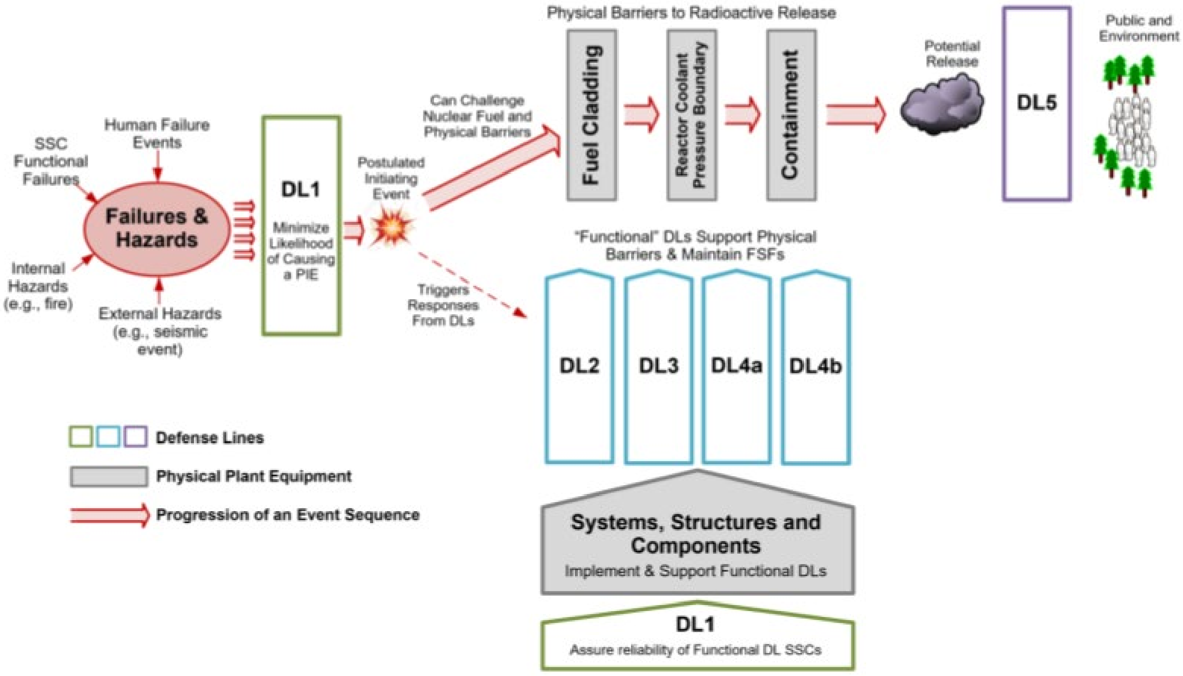

Figure 15.1-1: Defence-in-Depth Concept

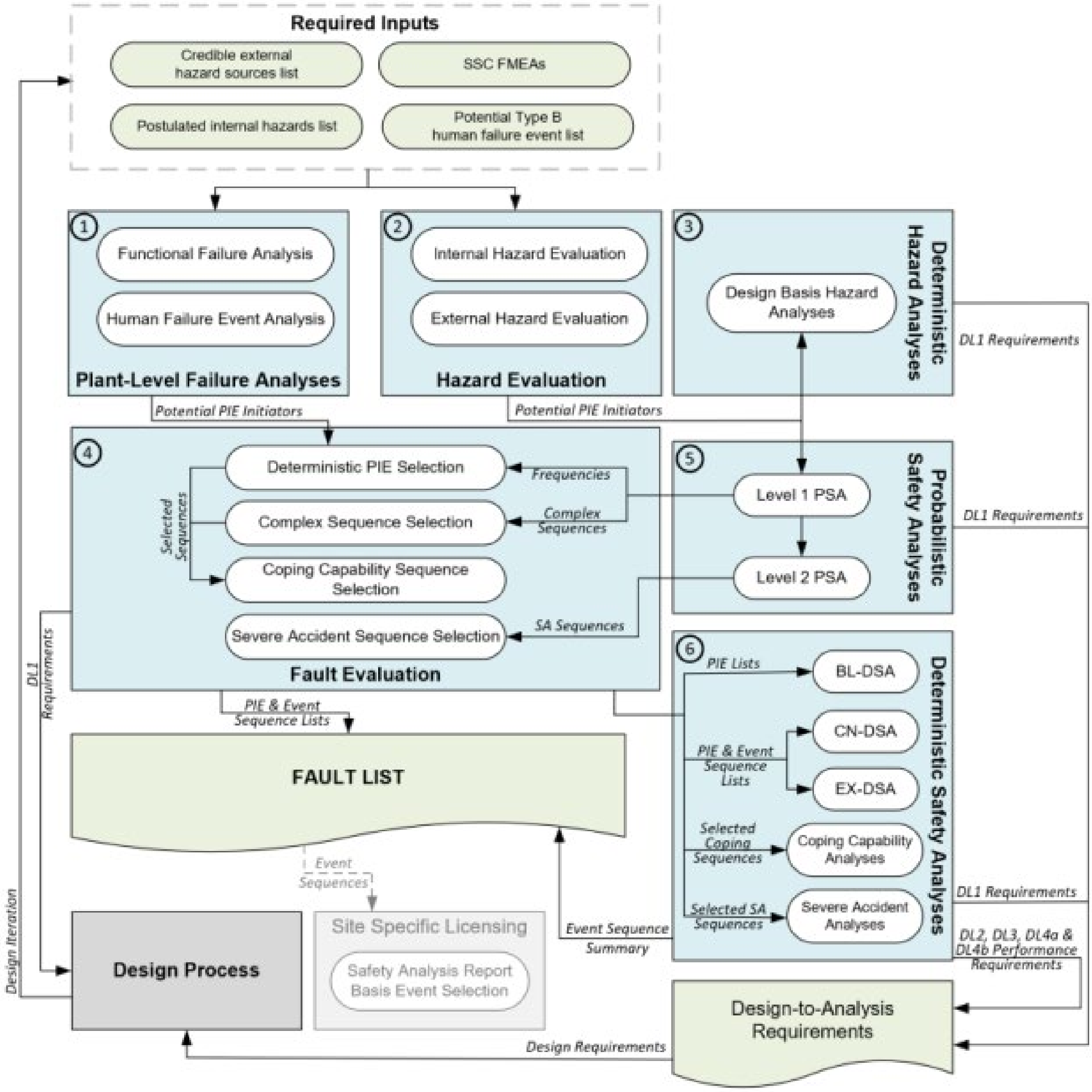

Figure 15.1-2: Safety Strategy Evaluation and Analysis Framework

15.1.9 References

15.1-1 NEDC-34180P, “BWRX-300 UK GDA Ch. 15.2: Safety Analysis - Identification, Categorisation, and Grouping of Postulated Initiating Events and Accident Scenarios,” Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

15.1-2 NEDC-34181P, “BWRX-300 UK GDA Ch. 15.3: Safety Analysis - Safety Objectives and Acceptance Criteria,” Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

15.1-3 NEDC-34182P, “BWRX-300 UK GDA Ch. 15.4: Safety Analysis - Human Actions,” Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

15.1-4 NEDC-34183P, “BWRX-300 UK GDA Ch. 15.5: Deterministic Safety Analyses, Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

15.1-5 NEDC-34184P, “BWRX-300 UK GDA Ch. 15.6: Probabilistic Safety Assessment,” Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

15.1-6 NEDC-34185P, “BWRX-300 UK GDA Ch. 15.7: Deterministic Safety Analysis – Analysis of Internal Hazards,” Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

15.1-7 NEDC-34186P, “BWRX-300 UK GDA Ch. 15.8: Safety Analysis - External Hazards,” Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

15.1-8 NEDC-34187P, “BWRX-300 UK GDA Ch. 15.9: Summary of Results of the Safety Analyses,” Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

15.1-9 IAEA, SSG-61, “Format and Content of the Safety Analysis Report for Nuclear Power Plants. Specific Safety Guide Series,” September 2021.

15.1-10 IAEA, SSR-2/1, “Safety of Nuclear Power Plants: Design Specific Safety Requirements,” (Revision 1).

15.1-11 NEDC-33934P, “BWRX-300 Safety Strategy,” Revision 1, GE-Hitachi Nuclear Energy, June 2024.

15.1-12 NEDC-34188P, “BWRX-300 UK GDA Ch. 16: Operational Limits and Conditions of Safe Operation,” Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

15.1-13 NEDC-34165P, “BWRX-300 UK GDA Ch. 3: Safety Objectives and Design Rules for SSCs,” Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

15.1-14 NEDC-34171P, “BWRX-300 UK GDA Ch. 9A: Auxiliary Systems,” Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

15.1-15 NEDC-34164P, “BWRX-300 UK GDA Ch. 2: Site Characteristics,” Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

15.1-16 NEDC-34190P, “BWRX-300 UK GDA Ch. 18: Human Factors Engineering,” Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

15.1-17 NEDC-34140P, “BWRX-300 UK GDA Safety Case Development Strategy,” GE-Hitachi Nuclear Energy, Americas, LLC.

15.1-18 NEDC-34274P, “BWRX-300 UK GDA Forward Action Plan Register,” GE-Hitachi Nuclear Energy, Americas, LLC.

APPENDIX A CLAIMS, ARGUMENTS AND EVIDENCE

The GDA CAE structure is defined within NEDC-34140P, "BWRX-300 UK GDA Safety Case Development Strategy," (SCDS) (Reference 15.1-17) 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".

APPENDIX B FORWARD ACTION PLAN

This subchapter does not directly support any forward actions, as defined within NEDC-34274P, “BWRX-300 UK GDA Forward Action Plan,” (Reference 15.1-18).