GE Hitachi Nuclear Energy

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BWRX-300 UK Generic Design Assessment (GDA) Chapter 15.3 – Safety Analysis – Safety Objectives and Acceptance Criteria

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EXECUTIVE SUMMARY

This document is Chapter 15.3, Safety Objectives and Acceptance Criteria, of the Preliminary Safety Report of the GEH BWRX-300 for the purposes of UK Generic Design Assessment. It presents the safety objectives and acceptance criteria for the Deterministic Safety Analysis and Probabilistic Safety Assessment.

ACRONYMS AND ABBREVIATIONS

Acronym	Explanation
ALARP	As Low As Reasonably Practicable
AOO	Anticipated Operational Occurrence
CFR	Code of Federal Regulation
DBA	Design Basis Accident
DEC	Design Extension Condition
DSA	Deterministic Safety Analysis
FAP	Forward Action Plan
GDA	Generic Design Assessment
GDC	General Design Criteria
GEH	GE Hitachi Nuclear Energy
IAEA	International Atomic Energy Agency
LOCA	Loss-of-Coolant Accident
LRF	Large Release Frequency
NPP	Nuclear Power Plant
PCT	Peak Cladding Temperature
PIE	Postulated Initiating Event
PSA	Probabilistic Safety Assessment
PSR	Preliminary Safety Report
SAFDL	Specified Acceptable Fuel Design Limit
SSCs	Structures, Systems, and Components
TEDE	Total Effective Dose Equivalent
UK	United Kingdom
U.S.	United States
USNRC	U.S. Nuclear Regulatory Commission

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Revision #	Section Modified	Revision Summary
A	All	Initial Revision

15.3 SAFETY OBJECTIVES AND ACCEPTANCE CRITERIA

Introduction

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 – NEDC-34179P, "BWRX-300 UK GDA Ch. 15.1: Safety Analysis - General Considerations," (Reference 15.3-1)
- 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 15.3-2)
- *PSR Ch. 15.3 – Safety Analysis - Objectives and Acceptance Criteria [this subchapter]*
- PSR Ch. 15.4 – NEDC-34182P, "BWRX-300 UK GDA Ch. 15.4: Safety Analysis - Human Actions," (Reference 15.3-3)
- PSR Ch. 15.5 – NEDC-34183P, "BWRX-300 UK GDA Ch. 15.5: Deterministic Safety Analysis," (Reference 15.3-4)
- PSR Ch. 15.6 – NEDC-34184P, "BWRX-300 UK GDA Ch. 15.6: Probabilistic Safety Assessment," (Reference 15.3-5)
- PSR Ch. 15.7 – NEDC-34185P, "BWRX-300 UK GDA Ch. 15.7: Deterministic Safety Analysis - Internal Hazards," (Reference 15.3-6)
- PSR Ch. 15.8 – NEDC-34186P, "BWRX-300 UK GDA Ch. 15.8: Safety Analysis - External Hazards," (Reference 15.3-7)
- PSR Ch. 15.9 – NEDC-34187P, "BWRX-300 UK GDA Ch. 15.9: Summary of Results of the Safety Analyses," (Reference 15.3-8)

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," (NPPs) (Reference 15.3-10) with the exception that Internal and External Hazards are discussed in two separate subchapters.

Subchapter Structure

This subchapter presents the acceptance criteria and safety objectives for the BWRX-300 Safety Analyses and comprises the following main sections:

- *15.3 – Introduction [this section]*
- 15.3.1 – Safety Objectives and Safety Analysis
- 15.3.2 – Deterministic Safety Analysis Acceptance Criteria
- 15.5.3 – Probabilistic Safety Assessment Acceptance Criteria
- 15.3.4 – Conclusion

Interfaces with other Chapters

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

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

Purpose

The purpose of this subchapter is to present the safety objectives and acceptance criteria used in the Deterministic Safety Analysis (DSA) and Probabilistic Safety Assessment (PSA) for the BWRX-300.

Scope

The scope of this subchapter comprises the safety analyses presented in NEDC-34183P (Reference 15.3-4) and NEDC-34184P (Reference 15.3-5).

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. GDA is an up-front, non-site-specific assessment of a generic NPP 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 employed in the safety argument. The demonstration of ALARP is typically achieved through the application of relevant good practice.

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 Ch. 15.3 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 15.3-3), NEDC-34183P (Reference 15.3-4), NEDC-34184P (Reference 15.3-5), NEDC-34185P (Reference 15.3-6) and NEDC-34186P (Reference 15.3-7). 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.

15.3.1 Safety Objectives and Safety Analysis

The DSA Anticipated Operational Occurrence (AOO) event sequences acceptance criteria are based on or derived from ensuring that the Specified Acceptable Fuel Design Limits (SAFDLs) are met for the following U.S. Nuclear Regulatory Commission (USNRC) 10 Code of Federal Regulations (CFR) Part 50, Appendix A General Design Criteria for Nuclear Power Plants," (GDC) (Reference 15.3-11):

- USNRC 10 CFR 50, Appendix A, GDC-10, "Reactor Design"
- USNRC 10 CFR 50, Appendix A, GDC-12, "Suppression of Reactor Power Oscillations"

- USNRC 10 CFR 50, Appendix A, GDC-17, "Electric Power Systems"
- USNRC 10 CFR 50, Appendix A, GDC-20, "Protection System Functions"
- USNRC 10 CFR 50, Appendix A, GDC-25, "Protection System Requirements for Reactivity Control Malfunctions"
- USNRC 10 CFR 50, Appendix A, GDC-26, "Reactivity Control System Redundancy and Capability"
- USNRC 10 CFR 50, Appendix A, GDC-33, "Reactor Coolant Makeup"
- USNRC 10 CFR 50, Appendix A, GDC-34, "Residual Heat Removal"

The DSA Design Basis Accident (DBA) Event Sequences acceptance criteria are based on or derived from ensuring that the 10 CFR 50.46(b), "Acceptance Criteria for Emergency Core Cooling Systems for Light-water Nuclear Power Reactors," (Reference 15.3-12) acceptance criteria for emergency core cooling systems are met.

Detailed consideration of the BWRX-300 Safety Objectives is presented within NEDC-34165P, "BWRX-300 UK GDA Ch. 3: Safety Objectives and Design Rules for Structures, Systems, and Components," (SSCs) (Reference 15.3-9).

15.3.2 Deterministic Safety Analysis Acceptance Criteria

The BWRX-300 design complies with the SAFDLs and 10 CFR 50.46(b) acceptance criteria.

Qualitative acceptance criteria are defined and met for each AOO and DBA to confirm the effectiveness of plant systems in maintaining the integrity of physical barriers against releases of radioactive material.

Derived qualitative and quantitative acceptance criteria are used to analyse AOOs or DBAs. Qualitative acceptance criteria are supported by experimental data, prescribed by regulatory requirements, or prescribed by applicable codes and standards. The results of the quantitative safety analysis confirm the derived acceptance criteria.

The acceptance criteria for Design Extension Conditions (DECs) are plant safety goals for Core Damage Frequency (CDF)and Large Release Frequency (LRF) that are specified in Table 15.5-3. The analysis results for DECs without core damage described in Section 15.5.5 may be conservatively compared to DBA acceptance criteria to demonstrate that the plant safety goals would not be exceeded.

Dose acceptance criteria for the Total Effective Dose Equivalent (TEDE) for an individual located at any point on the boundary of the exclusion zone are specified in: 10 CFR 50.34(a)(1)(ii)(D) "Contents of Applications; Technical Information,"(Reference 15.3-13), and 10 CFR 50, Appendix A, GDC 19 (Reference 15.3-10) for the control room personnel. NEDC-33934P, "BWRX-300 Safety Strategy," (Reference 15.3-14) requires that the calculated dose remains below limits established by national regulatory authorities applicable to a plant site; FAP items DSA-005 and DSA-006 pertain.

15.3.2.1 Acceptance Criteria for Analysis of Anticipated Operational Occurrences

The derived acceptance criteria for the DSA of AOOs are shown in Table 15.3-1.

15.3.2.2 Acceptance Criteria for Analysis of Design Basis Accidents

The derived acceptance criteria for the DSA of DBAs are shown in Table 15.3-2.

15.3.3 Probabilistic Safety Assessment Acceptance Criteria

The safety goals for the PSA of core damage frequency and large release frequencies are shown in Table 15.3-3 and are consistent with the USNRC Safety Goals for the Operations of Nuclear Power Plants; Policy Statement (Reference 15.3-15).

Table 15.3-1: Anticipated Operational Occurrence Deterministic Safety Analysis Acceptance Criteria

Fission Product Barrier or Fundamental Safety Function	Qualitative Acceptance Criteria	Quantitative Acceptance Criteria
General	An AOO will not escalate to a more serious accident condition unless other faults occur independently.	Not applicable.
	There is no consequential failure or loss of function of any fission product barrier.	Not applicable.
Fuel Rod	Fuel temperature results will not cause a loss of fuel rod mechanical integrity.	The calculated maximum fuel center temperature T_{center} remains below the fuel melting point T_{melt}.
	Fuel pellet-cladding results do not lead to loss of fuel rod mechanical integrity	The cladding strain acceptance criteria defined in Section 5.0 of Reference 15.3-16.
	Fuel cladding temperature results will not cause a fuel rod failure.	The calculated core Minimum Critical Power Ratio (MCPR) ensures that 99.9% of the fuel rods in the core are not susceptible to boiling transition during AOO events. With the reactor steam dome pressure less than 4.72 MPaG (685 psig), the calculated reactor thermal power is less than 25% of rated thermal power.
Reactor Coolant Pressure Boundary	Design conditions of the reactor coolant pressure boundary are not exceeded during the most severe pressurisation transient.	The calculated peak pressure associated with the reactor coolant pressure boundary does not exceed 110% of the design pressure or 11.38 MPaG (1650 psig).
	The reactor coolant pressure boundary maintains sufficient reactor coolant inventory for core cooling.	The calculated reactor water level is maintained at or above top of active fuel.
Primary Containment	Containment integrity is maintained. If an AOO results in an energy release to the containment, or loss of containment heat removal, then containment stresses (i.e., pressure and temperature) are limited such that there is no loss of a containment barrier safety function, and thus, the containment remains within its design limit values.	No AOOs result in a significant energy release to containment, or prolonged loss of normal containment cooling. The normal operation limits and conditions are applied to containment, and no AOO containment quantitative criteria is needed.
Long-Term Heat Removal	SSCs important for preserving the integrity of the reactor core and the containment can	Following AOO events that do not result in shutdown, a controlled condition is achieved.

Fission Product Barrier or Fundamental Safety Function	Qualitative Acceptance Criteria	Quantitative Acceptance Criteria
	remove residual heat for an extended period both during and after all applicable Postulated Initiating Events (PIE)s considered in all Operational States, including AOOs.	Following AOO events that require shutdown, the core remains shutdown independent of operator action or offsite support for at least 72 hours. For AOO events that rely on Defence Line 3 (DL3) mitigation for long-term cooling, the DL3 functions can provide cooling for at least 72 hours without operator action or offsite support.

Table 15.3-2: Design Basis Accident Acceptance Criteria

Fission Product Barrier or Fundamental Safety Function	Qualitative Acceptance Criteria	Quantitative Acceptance Criteria
General	Except for fuel cladding, there is no loss of function of any fission product barrier.	Not applicable
Fuel Rod	The number of fuel rod failures is conservatively estimated for DBAs.	The calculated number of failed rods does not result in exceeding the applicable radiological dose acceptance criteria.
	Mechanical fracturing of a fuel assembly under DBA loading conditions does not result in losing the ability to cool the fuel assembly.	The mechanical integrity of the fuel is established from the mechanical and thermal fuel analysis described in Section 4.2.2 of NEDC-34166P "BWRX-300 UK GDA Ch. 4: Reactor," (Reference 15.3-16).
Fuel Cooling	The calculated fuel cladding temperature is maintained at an acceptably low value and decay heat is removed for the extended period required by the long-lived radioactivity remaining in the core.	The calculated Peak Cladding Temperature (PCT) remains less than 1204°C (2200°F). The calculated total oxidation of the cladding nowhere exceeds 0.17 times the total cladding thickness before oxidation for DBAs where exceeding the oxidation thickness challenges the capability to cool the core.
Reactor Coolant Pressure Boundary	Design conditions of the reactor coolant pressure boundary are not exceeded during the most severe pressurisation transient because of a DBA.	The calculated peak pressure associated with the reactor coolant pressure boundary does not exceed 120% of the design pressure or 12.41 MPaG (1800 psig).
	The reactor coolant pressure boundary maintains sufficient reactor coolant inventory for core cooling	Conformance is demonstrated by meeting the fuel cooling and long-term heat removal criteria.
Primary Containment	Containment pressures and temperatures are maintained below the design values.	The calculated containment pressure does not exceed the design pressure 0.414 MPaG (60 psig). The calculated containment shell temperature does not exceed the design temperature 165.6°C (330°F).
	The local combustible gas concentrations in the containment are within the range where deflagration or detonation cannot occur.	Containment atmosphere remains sufficiently mixed such that deflagration or detonation thresholds are not exceeded.
	Containment capability will be retained to reduce the containment pressure and temperature following a DBA to minimise the release of fission products to the	The calculated containment pressure reduces to less than 50% of the calculated peak pressure for the most limiting Loss-of-Coolant Accident (LOCA) within 24 hours.

Fission Product Barrier or Fundamental Safety Function	Qualitative Acceptance Criteria	Quantitative Acceptance Criteria
	environment and to preserve containment integrity and leak tightness.	
Reactivity Control	Reactivity control required to bring the reactor to cold shutdown is maintained.	Shutdown margin is established to assure that the reactor can be brought subcritical with the highest-worth control rod pair withdrawn when the core is in its most reactive condition. The subcriticality value is 0.38% $\Delta k/k$ with the highest-worth control rod pair analytically determined.
Long-Term Heat Removal	SSCs important for preserving the integrity of the reactor core and the containment are capable of removing residual heat for an extended period both during and after all applicable PIEs considered in all operational states, and DBAs.	Long-term cooling is maintained for a minimum of 72 hours independent of operator action and offsite support, and for 30 days with credit for operator actions and on-site resources. For DBA events that result in shutdown, the plant can achieve and maintain safe-shutdown conditions with the average reactor coolant temperature below 215.6°C (420°F).

Table 15.3-3: Probabilistic Safety Goals

Qualitative Acceptance Criteria	Quantitative Acceptance Criteria
Core damage frequency	The sum of frequencies of all event sequences that can lead to significant core degradation is less than 1E-5 per reactor year
Large release frequency	The calculated sum of frequencies of all event sequences that can lead to any release to the environment that requires long term relocation of the location population is less than 1E-6 per reactor year.

15.3.4 References

15.3-1 NEDC-34179P, “BWRX-300 UK GDA Ch. 15.1: Safety Analysis - General Considerations,” Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

15.3-2 NEDC-34180P, “BWRX-300 UK GDA Ch. 15.2: Safety Analysis - Identification, Categorization, and Grouping of Postulated Initiating Events and Accident Scenarios,” Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

15.3-3 NEDC-34182P, “BWRX-300 UK GDA Ch. 15.4: Safety Analysis - Human Actions,” Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

15.3-4 NEDC-34183P, “BWRX-300 UK GDA Ch. 15.5: Deterministic Safety Analysis,” Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

15.3-5 NEDC-34184P, “BWRX-300 UK GDA Ch. 15.6: Probabilistic Safety Assessment,” Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

15.3-6 NEDC-34185P, “BWRX-300 UK GDA Ch. 15.7: Deterministic Safety Analyses - Internal Hazards,” Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

15.3-7 NEDC-34186P, “BWRX-300 UK GDA Ch. 15.8: Safety Analysis - External Hazards,” Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

15.3-8 NEDC-34187P, “BWRX-300 UK GDA Ch 15.9: Summary of Results of the Safety Analyses,” Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

15.3-9 NEDC-34165P, “BWRX-300 UK GDA Ch. 3: Safety Objectives and Design Rules for SSCs,” Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

15.3-10 IAEA, SSG-61, “Format and Content of the Safety Analysis Report for Nuclear Power Plants. Specific Safety Guide,” September 2021.

15.3-11 Appendix A to Part 50 – General Design Criteria for Nuclear Power Plants. USNRC. 10 CFR.

15.3-12 Acceptance Criteria for Emergency Core Cooling Systems for Light-water Nuclear Power Reactors. USNRC. 10 CFR 50.46.

15.3-13 Contents of Applications; Technical Information. USNRC. 10 CFR 50.34.

15.3-14 NEDC-33934P, “BWRX-300 Safety Strategy,” Revision 1, GE-Hitachi Nuclear Energy.

15.3-15 Safety Goals for the Operations of Nuclear Power Plants; Policy Statement. Federal Register, USNRC, Volume 51, No. 149, pp. 28044-28049, August 4, 1986. (Corrected and reprinted at Federal Register, Volume 51, No. 162, pp. 30028-30033, August 21, 1986).

15.3-16 NEDC-34166P, “BWRX-300 UK GDA Ch. 4: Reactor (Fuel and Core),” Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

15.3-17 NEDC-34140P, “BWRX-300 UK GDA Safety Case Development Strategy,” Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

15.3-18 NEDC-34274P, “BWRX-300 UK GDA Forward Action Plan,” 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 15.3-17) 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 chapter does not directly support any forward actions, as defined within NEDC-34274P, ‘BWRX-300 UK GDA Forward Action Plan,” (Reference 15.3-18).