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BWRX-300 UK Generic Design Assessment (GDA) Chapter 22 - Structural Integrity of Metallic Systems, Structures and Components

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EXECUTIVE SUMMARY

This purpose of this chapter is to justify the structural integrity of metallic Structures, Systems, and Components (SSCs) of the United Kingdom (UK) BWRX-300 design and describe how the risk of structural failure may be demonstrated to be As Low As Reasonably Practicable (ALARP).

Structural reliability of SSCs will be justified according to the consequences of their failure as established by a system of component structural integrity classification, which introduces additional requirements for components classified as High Integrity that are over and above Safety Class 1 (SC1) SSCs. This classification system was previously used for the UK Advanced Boiling Water Reactor (ABWR) Generic Design Assessment (GDA) submission.

This chapter defines which SSCs are structural integrity related and presents a level of detail commensurate with a Step 2 GDA.

The scope of the structural integrity case comprises safety significant metallic SSCs, as identified by their classification, with respect to their structural integrity for conditions that may credibly occur during a 60-year period of operation.

System interfaces/dependencies are identified, and suitable cross references are used to direct the reader to the relevant interfacing chapters of the safety justification.

Claims and arguments relevant to GDA Step 2 objectives and scope are summarised in Appendix A.

Appendix B has been included to capture Forward Action Plan (FAP) items, where a single FAP item has been identified.

ACRONYMS AND ABBREVIATIONS

Acronym	Explanation
ABWR	Advanced Boiling Water Reactor
ALARP	As Low As Reasonably Practicable
AOO	Anticipated Operational Occurrence
ASME	American Society of Mechanical Engineers
BPVC	Boiler and Pressure Vessel Code
BWR	Boiling Water Reactor
CAE	Claims, Arguments and Evidence
DBA	Design Basis Accident
DEC	Design Extension Condition
EC	Erosion Corrosion
ENIQ	European Network for Inspection and Qualification
ESBWR	Economic Simplified Boiling Water Reactor
FAC	Flow Accelerated Corrosion
FAP	Forward Action Plan
FSF	Fundamental Safety Function
GDA	Generic Design Assessment
GEH	GE Hitachi Nuclear Energy
HI	High Integrity
IAEA	International Atomic Energy Agency
IASCC	Irradiation Assisted Stress Corrosion Cracking
IGSCC	Intergranular Stress Corrosion Cracking
IoF	Incredibility of Failure
ISI	In-service Inspection
LfE	Learning from Experience
NDT	Non-Destructive Testing
ONR	Office for Nuclear Regulation
OPEX	Operating Experience
PCER	Pre-Construction Environmental Report
PCSR	Pre-Construction Safety Report
PSI	Preservice Inspection
PSR	Preliminary Safety Report
QA	Quality Assurance
QEDS	Qualified Examination Defect Size
RCPB	Reactor Coolant Pressure Boundary
RGP	Relevant Good Practice

Acronym	Explanation
RIV	Reactor Isolation Valve
RPV	Reactor Pressure Vessel
SAPs	Safety Assessment Principles
SC	Safety Class
SC1	Safety Class 1
SC2	Safety Class 2
SC3	Safety Class 3
SCC	Stress Corrosion Cracking
SCDS	Safety Case Development Strategy
SCN	Non-Safety Class
SFC	Safety Functional Claim
SSCs	Structures, Systems, and Components
TAGSI	Technical Advisory Group on Structural Integrity
UK	United Kingdom
USNRC	U.S. Nuclear Regulatory Commission

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None.

REVISION SUMMARY

Revision #	Section Modified	Revision Summary
A	All	Initial Issuance
B	All	Update for end of GDA Step 2 consolidation

22 STRUCTURAL INTEGRITY OF METALLIC SYSTEMS, STRUCTURES AND COMPONENTS

Introduction

The BWRX-300 leverages the United States Nuclear Regulatory Commission (USNRC) approved Economic Simplified Boiling Water Reactor (ESBWR) design, proven in-use materials, off-the-shelf components, and design pressures and temperatures within the range of the existing Boiling Water Reactor (BWR) design and experience base, detailed in 005N9751, "BWRX-300 General Description," (Reference 22-1). Many of the components utilised in the BWRX-300 have significant Operating Experience (OPEX) in the nuclear and power industries along with an existing supply chain, which minimises risk. The risks minimized include removal of uncertainty in manufacturing, material behaviour, testing, Quality Assurance (QA), and acceptance by the USNRC and various codes.

This chapter describes the top-level safety case to demonstrate the structural integrity of the metallic Structures, Systems, and Components (SSCs) of the United Kingdom (UK) BWRX-300. It presents how the claims and arguments presented in Appendix A are organized and used to demonstrate an adequate level of structural integrity commensurate with the required level of structural reliability and consequence of failure.

The level of structural integrity is determined by a structural integrity classification scheme similar to that used in the previous Generic Design Assessment (GDA) for the Advanced Boiling Water Reactor (ABWR) in GA91-9101-0101-08000, "UK ABWR Generic Design Assessment, Generic Pre-Construction Safety Report (PCSR) Chapter 8: Structural Integrity," (Reference 22-2). This scheme sub-divides the Safety Class 1 (SC1) components identified by the GEH safety classification system in 005N9461, "BWRX-300 Structures, Systems, and Components Safety Classification," (Reference 22-3) into High Integrity (HI) and SC1 components. HI components are identified as components where failure is intolerable and for which no physical protection is provided, or protection is not reasonably practicable. For these components, the safety arguments are presented in an approach consistent with that suggested by the UK Technical Advisory Group on Structural Integrity (TAGSI) in "The Demonstration of Incredibility of Failure in Structural Integrity Safety Cases," (Reference 22-4) and in line with good practice. These arguments are enhanced by additional measures for defect tolerance and the application of qualified inspection based on the European Network for Inspection and Qualification (ENIQ). For components where the failure consequences are less severe, (i.e., Safety Classes 1, 2 and 3), the arguments are presented to provide compliance with appropriate codes and standards.

Claims, Arguments and Evidence

All functions have been derived and substantiated taking into account Relevant Good Practice (RGP) and OPEX, and processes are in place to maintain these through-life (Claim 2.1):

- The design of the system/structure has been substantiated to achieve the safety functions in all relevant operating modes (Sub-Claim 2.1.2).
- The system/structure design has been undertaken in accordance with relevant design codes and standards (RGP) and design safety principles and taking account of OPEX to support reducing risks ALARP (Sub-Claim 2.1.3).
- System/structure performance will be validated by suitable testing throughout manufacturing, construction, and commissioning (Sub-Claim 2.1.4).
- Ageing and degradation mechanisms will be identified and assessed in the design. Suitable examination, inspection, maintenance, and testing will be specified to maintain systems/structures fit-for-purpose through-life (Sub-Claim 2.1.5).

- The BWRX will be designed so that it can be decommissioned safely, using currently available technologies, and with minimal impact on the environment and people (Sub-Claim 2.1.6).

Safety risks have been reduced as low as reasonably practicable (Claim 2.4):

- RGP has been taken into account across all disciplines (Sub-Claim 2.4.1).
- OPEX and Learning from Experience (LfE) has been taken into account across all disciplines (Sub-Claim 2.4.2).
- Optioneering (all reasonably practicable measures have been implemented to reduce risk) (Sub-Claim 2.4.3).

Purpose

This purpose of this chapter is to justify the structural integrity of metallic SSCs of the BWRX-300 design and describe how the risk of structural failure may be demonstrated to be ALARP.

Structural reliability of SSCs will be justified according to the consequences of their failure as established by a system of component structural integrity classification, which introduces additional requirements for components classified as High Integrity (or Incredibility of Failure (IoF)), that are over and above SC1 SSCs. This classification system was previously used for the UK ABWR GDA submission.

This chapter defines which SSCs are structural integrity related and presents a level of detail commensurate with a Step 2 GDA (claims and arguments only).

Specifically, the objectives of this chapter and its supporting documents are to:

- Identify relevant codes and standards that form the structural integrity requirements
- Identify the structural integrity safety functions and specify the safety classifications of the SSCs that are within the scope of this chapter
- Specify the relevant Safety Functional Claims (SFCs) related to the structural integrity topic area
- Identify links to other chapters of the Preliminary Safety Report (PSR) to ensure consistency within the structural integrity topic area across the whole safety case

Scope

The scope of the structural integrity case comprises safety significant metallic SSCs, as identified by their classification, with respect to their structural integrity for all conditions that may credibly occur during a 60-year period of operation. The decommissioning period is also considered at a strategic level during GDA (see NEDO-34193, “BWRX-300 UK GDA Chapter 21: Decommissioning and End of Life Aspects,” (Reference 22-5)), and this chapter presents the work done in this area.

The approach to substantiate structural integrity is summarised as follows:

- Identify and categorise Safety Functional Requirements according to their importance to safety
- Identify SSCs that deliver each safety function
- Establish suitably rigorous requirements for design, construction, and operation, according to classification

This chapter describes the claims associated with the substantiation of the structural integrity of safety significant metallic pressure boundary components and their supports. This includes

HI and SC1, Safety Class 2 (SC2), and Safety Class 3 (SC3) components. Structural integrity encompasses a number of technical areas including metallurgy, material properties and testing, ageing and degradation mechanisms, welding engineering, stress analysis, fracture mechanics, and Non-Destructive Testing (NDT) techniques.

The design and safety requirement compliance of structural integrity related SSCs are also covered by the relevant PSR chapter.

International Atomic Energy Agency (IAEA) SSG-61, “Format and Content of the Safety Analysis Report for Nuclear Power Plants,” (Reference 22-6) does not include ‘Structural Integrity’ as a chapter, and thus does not provide chapter structure and content guidance. The chapter structure and contents take note of previous (UK-specific) GDA work for the UK ABWR, GA91-9101-0101-08000 (Reference 22-2), on this topic.

It is noted that structural integrity concentrates on the integrity of static safety significant components and structures. Mechanical systems like lifting equipment, the fuel handling machine, pressure relief systems, pumps, valves, and ventilation systems are in the mechanical engineering scope and are not considered within the scope of this chapter. Civil structures, electrical systems, and instrumentation and control systems are also out of the scope of this chapter.

Interfaces with Other Chapters

This chapter also provides links to other key chapters within the safety case that form part of or link to the case for this topic area.

- NEDO-34165, “BWRX-300 UK GDA Chapter 3: Safety Objectives and design Rules for SSCs,” (Reference 22-7) establishes the principles for the safety classification of SSCs. The design requirements for (in particular) Safety Class 1 SSCs are an input to the design principles for HI components defined in this chapter.
- NEDO-34167, “BWRX-300 UK GDA Chapter 5: Reactor Coolant System and Associated Systems,” (Reference 22-8) provides the system description of the Reactor Pressure Vessel (RPV), the reactor internals, control rod mechanisms, Main Steam System (including main steam isolation valves), the Feedwater System, and core cooling systems. The principles of design are established in this chapter.
- NEDO-34168, “BWRX-300 UK GDA Chapter 6: Engineered Safety Features,” (Reference 22-9) provides the system description of the Isolation Condenser System, Containment and Associated Systems, and Control Room Habitability.
- NEDO-34171, “BWRX-300 UK GDA Chapter 9A: Auxiliary Systems,” (Reference 22-10) provides the system description of the cooling water systems (e.g., ultimate heat sink and reactor building cooling water). Although there are no HI SSCs on the systems described, the metallic SSCs will be designed against the principles established in this chapter.
- NEDO-34173, “BWRX-300 UK GDA Chapter 10: Steam and Power Conversion Systems,” (Reference 22-11) provides the system description for the Turbine Generator, Turbine Main Steam, Turbine Auxiliary Steam and Turbine Bypass System, Extraction Steam System, Turbine Gland Steam System, Feedwater Heater Drain and Vent System, Condenser, Circulating Water System, Condensate and Feedwater System, and Condensate Purification System. Although there are no HI SSCs on the systems described, the metallic SSCs will be designed against the principles established in this chapter.
- NEDO-34175, “BWRX-300 UK GDA Chapter 12: Radiation Protection,” (Reference 22-12) - Radiation environment affects structural integrity related

considerations such as material selection, through-life ageing, and through-life inspection.

- NEDO-34176, “BWRX-300 UK GDA Chapter 13: Conduct of Operations,” (Reference 22-13) - It is necessary to understand the plant’s proposed operating regime because it affects material selection and through-life inspections.
- NEDO-34184, “BWRX-300 UK GDA Chapter 15.6: Probabilistic Safety Assessment,” (Reference 22-14) and NEDO-34187, “BWRX-300 UK GDA Chapter 15.9: Summary of Results of the Safety Analyses,” (Reference 22-15) - Component reliability is an input to the Probabilistic Safety Assessment work, which is partly affected by structural integrity assessment work.
- NEDO-34188, “BWRX-300 UK GDA Chapter 16: Operational Limits and Conditions of Safe Operation,” (Reference 22-16) - Operating limits and conditions can affect material selection, through-life ageing and degradation, and through-life maintenance/inspection of structural integrity related SSCs.
- NEDO-34190, “BWRX-300 UK GDA Chapter 18: Human Factors Engineering,” (Reference 22-17) - The plant is designed to facilitate the through-life maintenance and inspection of structural integrity related SSCs, which requires Human Factors considerations.
- NEDO-34193, “BWRX-300 UK GDA Chapter 21: Decommissioning and End of Life Aspects,” (Reference 22-5) discusses general requirements for decommissioning of SSCs.
- NEDO-34195, “BWRX-300 UK GDA Chapter 23: Reactor Chemistry,” (Reference 22-18) - The plant’s water chemistry regime affects material selection, ageing and degradation mechanisms, and through-life maintenance and inspection considerations.
- NEDO-34197, “BWRX-300 UK GDA Chapter 27: ALARP Evaluation,” (Reference 22-19) - The structural integrity related SSC design and operations are obliged to demonstrate that they have reduced risks ALARP. This includes demonstrating compliance with relevant codes and standards (RGP) and demonstrating that appropriate OPEX has been taken into account.

Note: PSR Chapters 5, 6, 9A, and 10 should be referred to for design summaries of SSCs.

Claims and arguments relevant to GDA step 2 objectives and scope are summarised in Appendix A.

Appendix B has been included to capture any Forward Action Plan (FAP) items.

22.1 Safety Classification

The overall safety philosophy for the design of the BWRX-300 is referred to as the Safety Strategy and is presented in 006N5064, "BWRX-300 Safety Strategy," (Reference 22-20). The objective of the Safety Strategy is to establish a design with a high level of safety. This is accomplished through incorporation of design requirements based on the principles set forth in IAEA SSR-2/1, "Safety of Nuclear Power Plants: Design," (Reference 22-21).

The BWRX-300 approach to categorisation of functions and classifying SSCs is consistent with IAEA SSR-2/1 (Reference 22-21) and IAEA SSG-30, "Safety Classification of Structures, Systems and Components in Nuclear Power Plants," (Reference 22-22). As such, this approach is based primarily on deterministic methods in that the classifications are not based on the calculated risk of each component but are assigned deterministically based on the functions performed by the SSC and the importance of those functions. The overall approach is to identify functions that affect nuclear safety, assign a safety category to these functions based on their importance, and then assign a safety class to the components that perform those functions.

The assignment of safety class (SC1, SC2 and SC3) is described in PSR Chapter 3 (Reference 22-7). In addition, the following component classifications are made for components that perform Fundamental Safety Functions (FSFs) but may not be explicitly defined as part of a defense line function.

- Components that are part of design provisions that perform a FSF, whose failure is considered "practically eliminated" are assigned to SC1, e.g., the RPV.
- Components that make up the fission product barriers (i.e., fuel cladding, Reactor Coolant Pressure Boundary (RCPB) and containment) are assigned to SC1.
- Components that are part of the RCPB as defined below (definition taken from 10 CFR 50.2) are assigned to SC1.

The RCPB includes the pressure-retaining components of boiling and pressurized water-cooled nuclear power reactors, such as pressure vessels, piping, pumps, and valves, which are:

- Part of the reactor coolant system
- Connected to the reactor coolant system, up to and including any of the following:
    - The outermost containment isolation valve in system piping which penetrates primary reactor containment.
    - The second of two valves normally closed during normal reactor operation in system piping which does not penetrate primary reactor containment.
    - The reactor coolant system safety and relief valves.

The FSFs for the BWRX-300 align with IAEA SSR 2/1 and are as follows:

- Control of reactivity
- Removal of heat from the fuel (in the reactor, during fuel storage and handling, and including long-term heat removal)
- Confinement of radioactive material, shielding against radiation, and control of planned radioactive releases, as well as limitation of accidental radioactive releases

The system functional requirements for the reactor coolant system are described in PSR Chapter 5 (Reference 22-8) and in NEDC-34272P, "BWRX-300 UK GDA Topic Report – Reactor Pressure Vessel Structural Integrity Substantiation Methodology," (Reference 22-23).

22.1.1 Structural Integrity Classification

The approach to classifying SSCs is consistent with IAEA SSR 2/1 (Reference 22-21) and IAEA SSG-30 (Reference 22-22). The classification is conducted to identify the importance of the SSCs with respect to safety. The methodology for classification is described in Section 3.2 of PSR Chapter 3 (Reference 22-7) in accordance with:

- Safety Class
- Seismic Category
- Quality Group

Table 3-3 of PSR Chapter 3 (Reference 22-7) tabulates the design and fabrication requirements for each Quality Group.

The frequency and consequences of failure of SC1 components vary significantly. As the risk of failure varies, so does the required assurance of structural integrity. In order to identify where the very highest standards of structural integrity should apply, GA91-9201-0003-00054 (RD-GD-0001), “Structural Integrity Classification Procedure,” (Reference 22-24) describes a refined scheme of classification to be adopted which sub-divides SC1 into HI and SC1.

Table 22-1 shows structural integrity classes and illustrates the criteria in terms of consequences of failure. HI is assigned where failure can lead to severe core damage, but where a single line of protection exists; generally, this means that effective containment exists to limit the offsite consequences to a tolerable level.

The classification of HI components follows the process described in GA91-9201-0003-00054 (RD-GD-0001) (Reference 22-24). On the basis of this process, UK ABWR GDA experience has suggested that the RPV (Cylindrical Shell, Top Head/Bottom Head, Nozzles as part of the vessel), Reactor Isolation Valves (RIVs) and any connecting welds within the containment vessel are likely to be selected as HI components. Details of the RPV, in relation to structural integrity safety requirements, are presented in NEDC-34272P (Reference 22-23).

22.2 Structural Integrity Claims and Arguments

The structural integrity claims are described in Appendix A. These claims are used to demonstrate suitably robust structural integrity for Safety Class 1, 2, 3, and HI SSCs of the BWRX-300 through a series of arguments which are appropriate for that class.

The standards by which structural integrity is assured reflects the functional reliability requirements of the SSCs commensurate with their safety classification. The structural integrity of SC1 and HI SSCs will, at a minimum, be justified by evidence of compliance with the requirements of well-established and appropriate design codes.

The structure of the topic reports for HI components is consistent with that recommended by the UK TAGSI. This provides an approach for justification of high structural reliability claims by establishing diverse evidence of conceptual Defence-in-Depth against the risk of failure. At the highest level, the safety case for HI components is structured accordingly to meet the sub-claims as follows:

- Structural Integrity is assured by good design and taking into account relevant BWRX-300 OPEX (Sub-Claim 2.1.3)
- Structural Integrity is assured by material selection and quality manufacturing (Sub-Claims 2.1.4 and 2.1.5)
- Functional testing provides a demonstration of integrity at start of life (Sub-Claim 2.1.4)
- Through-life integrity is demonstrated by analysis and inspection (Sub-Claims 2.1.3 and 2.1.5)
- Inspection and monitoring regularly validate integrity through-life (Sub-Claim 2.1.5)

Each claim is supported by a series of arguments, which will each be substantiated in GDA Step 3 by identification of robust and diverse evidence, typically compiled as a dossier of technical information, data, and analyses reports.

This chapter presents the methodology for developing structural integrity safety cases for each of the safety classes for metal components and structures.

22.2.1 High Integrity Components

Structural integrity safety cases (topic reports) for HI components will be developed in accordance with previous practice for the ABWR GDA, which is consistent with the recommendations of TAGSI. NEDC-34272P (Reference 22-23) describes the methodologies and processes used to substantiate the structural integrity case for the RPV.

The failure of HI components can lead to radiological consequences, but the process of structural integrity classification will identify evidence that effective containment exists to limit the offsite consequences. It is necessary that the structural integrity of the HI regions is substantiated to a higher degree of rigour than that required for SC1 components. This is provided by evidence to demonstrate that welds will be subject to qualified manufacturing inspections, supported by an elastic-plastic fracture assessment to demonstrate tolerance to defects as described for HI components in Sections 22.2.4 and 22.2.7.

Sub-Claims 2.1.3 and 2.1.4 are intended to establish high quality through good design and manufacture, supplemented by Sub-Claim 2.1.4 that functional testing will be used to demonstrate fitness for purpose at start of life. This is the foundation for demonstration of very high reliability through the avoidance of significant defects.

Sub-Claim 2.1.5 states that measures are in place to demonstrate that HI components are tolerant to through-life degradation. This is demonstrated by the results of assessments of through-life crack growth to show that such mechanisms will not threaten integrity over a specific interval. This exceeds conventional design code requirements to provide a further demonstration of integrity by acknowledging that defects may be present and demonstrating tolerance to them.

In order to support Sub-Claim 2.1.3, there are nine arguments identified at this stage to demonstrate the highest reliability of components (further information is provided in NEDC-34272P (Reference 22-23)).

The arguments are structured around design, analysis, avoidance of fracture, material selection, manufacture, inspection, testing, forewarning of failure, and OPEX.

22.2.2 Design

To achieve a high quality of build, HI components comply with the requirements of relevant and widely used nuclear codes and standards. The relevant design codes and standards for HI components are presented in PSR Chapter 3 (Reference 22-7). Additional measures exceeding the requirements of these codes are identified and implemented as follows:

- The appropriate fracture toughness test for an HI component will be determined and implemented at the stage of product manufacture
- Inspection qualification according to ENIQ methodology in GA91-9201-0003-00057 (G-TY-53082), "Inspection Qualification Strategy," (Reference 22-25) will be used to achieve the reliability of objective based manufacturing NDT
- Defect tolerance is substantiated by defect tolerance assessment, following "Assessment of the Integrity of Structures Containing Defects," (Reference 22-26)
- Independent third-party inspection

During the design stage, the potential in-service ageing and degradation of the components and previous OPEX are considered. Novel design is avoided or adequately justified.

22.2.3 Design Analysis

In order to demonstrate that HI components are designed in compliance with allowable stress limits, fatigue usage factors and fast fracture limits as specified in the design codes (see Section 22.2.11) and the following failure modes will be assessed:

- Excessive deformation and plastic instability
- Buckling
- Progressive deformation (ratchetting)
- Fatigue (initiation and crack growth)
- Fast fracture

Details of any stress analysis to be used for design substantiation will be provided in GDA Step 3.

22.2.4 Avoidance of Fracture

A limited set of HI regions will be selected for R6 defect tolerance assessment in GDA Step 3. UK ABWR GDA experience has suggested that it is likely the RPV will be selected as an HI component. This is demonstrated by the results of assessments of through-life crack growth to show that such mechanisms will not threaten integrity over a specific interval. This exceeds conventional design code requirements to provide a further demonstration of integrity by acknowledging that defects may be present and demonstrating their tolerance. Defect tolerance is demonstrated by fracture assessments to establish tolerance to defects smaller than a Qualified Examination Defect Size (QEDS) by a size margin of two. For HI components, the elastic-plastic fracture mechanics methodology of the R6 procedure will be used for defect tolerance assessment, as described in GA91-9201-0003-00056 (RD-GD-0003), "Defect

Tolerance Assessment Plan," (Reference 22-27). Evidence is provided to identify how pressure-temperature limits are prescribed and controlled to prevent rupture, particularly at low temperatures during operation. The results of the defect tolerance assessments establish a QEDS for each region subject to assessment. Inspection qualification, conducted in accordance with ENIQ methodology, will be applied to confidently establish capability of detection for defects equal to or larger than the QEDS.

22.2.5 Material Selection

The material and process control requirements for the BWRX-300 components are defined in 006N5956, "Materials and Process Controls," (Reference 22-28) to ensure the reliability of the plant operations through the design life by minimizing irradiation of the plant components, corrodents, and mitigating the degradation of materials, especially from Intergranular Stress Corrosion Cracking (IGSCC) through material chemistry, heat treatment, contamination, and material processes controls.

Materials are selected to satisfy the design requirements for components to ensure that they perform safely throughout the design lifetime of the plant. In addition, previous OPEX from the UK ABWR has been applied. In principle, carbon steels including atmospheric corrosion resistant steels and low alloy steels are used as basic materials. In order to minimize levels of radiation from corrosion products, reactor internals are made of austenitic stainless steel. Materials selection and controls address material degradation issues in the reactor system such as Stress Corrosion Cracking (SCC), general corrosion, and flow accelerated corrosion.

SCC is considered the dominant form of corrosion damage in a BWR. Significant efforts through the years have been expended to understand it and control it. The different degradation mechanisms that potentially affect the integrity of the construction materials that are used in the BWRX-300 are discussed in Sections 5.2.1 and 5.2.2 of PSR Chapter 5 (Reference 22-8).

22.2.6 Manufacture

The manufacturing of components will be performed in accordance with approved procedures. Novel approaches such as electron beam welding and Powdered Metal – Hot Isostatic Pressing will only be utilized for the BWRX-300 once they are proven and accepted by regulators.

For the susceptible austenitic stainless steels or Nickel base alloys that are used for construction, process controls are exercised during various stages of component manufacturing and reactor construction to avoid fabrication-induced stresses that could lead to stress corrosion crack initiation. These processing steps can introduce surface cold work or localized sensitization. The processes that need to be controlled include bending or forming, final machining, grinding, polishing, and welding.

Fabrication requirements for the RPV are described in 006N7441, "BWRX-300 Reactor Pressure Vessel Fabrication Requirements," (Reference 22-29) and are presented in NEDC-34272P (Reference 22-23).

22.2.7 Inspection

The results of the defect tolerance assessments establish a QEDS for each region subject to assessment. Inspection qualification, conducted in accordance with ENIQ methodology, will be applied to confidently establish capability of detection for defects equal to or larger than the QEDS. The substantiation for HI components will describe R6 defect tolerance assessment in detail in GDA Step 3. Supplemental measures will be applied to support these claims for HI components. These will include additional fracture toughness testing to directly characterise the fracture toughness of the material in order to inform the defect tolerance assessment and to meet Sub-Claim 2.1.4. For some HI components, additional stringent control of chemical composition will be specified to minimise the effect of degradation mechanisms such as irradiation embrittlement or thermal ageing.

To ensure high reliability, manufacturing inspections will be applied to the HI components, for which the inspection system including procedure, equipment, and personnel, will be qualified according to the ENIQ-based methodology for qualification of NDT. The approach for inspection qualification is described in GA91-9201-0003-00057 (G-TY-53082) (Reference 22-25). These qualified inspections provide high confidence in establishing the absence of significant defects at the end of the manufacturing and at the Preservice Inspection (PSI) stages. The output of the PSI, completed before the start of operation, is the generation of a set of benchmark data against which future In-service Inspection (ISI) results can be compared.

22.2.8 Testing

The Reactor Coolant System (RCS) (see PSR Chapter 5, (Reference 22-8)) is designed with provisions for initial and periodic testing of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC) Class 1 and 2 equipment, including hydrostatic testing conducted in accordance with requirements of the ASME BPVC and Regulatory Guide 1.22, “Periodic Testing of Protection System Actuation Functions.”

Hydrostatic pressure testing is conducted on pressure vessels, pipework, and systems after completion of manufacture and after installation according to the requirements of the ASME BPVC to confirm integrity at the start of life through verifying strength and leak tightness of relevant SSCs.

Further information for the testing of SSCs can be found in PSR Chapter 5 (Reference 22-8), PSR Chapter 6 (Reference 22-9), PSR Chapter 9A (Reference 22-10), and PSR Chapter 10 (Reference 22-11).

Detailed records will be obtained at the time of testing to provide as evidence to demonstrate compliance with the appropriate requirements of the design code (beyond GDA Step 2).

22.2.9 Forewarning of Failure

In order to support Sub-Claim 2.1.5, early indication of degradation to prompt corrective action before gross failure occurs, arguments are identified to demonstrate that effective systems are in place to provide forewarning of failure which includes ISI, irradiation surveillance, monitoring of plant transients, and leak detection.

This is achieved by specification of ISI in accordance with 006N6279, “BWRX-300 In Service Inspection Requirements,” (Reference 22-30) to detect any degradation in good time before defect growth could significantly compromise structural integrity. ISI is also used to periodically confirm the absence of unanticipated degradation. ISI is a particularly important provision to forewarn of failure, and it will be specified for the HI locations in accordance with the robust ENIQ-based qualification methodology as described in GA91-9201-0003-00057 (G-TY-53082) (Reference 22-25). More generally, ISI will be specified in accordance with ASME BPVC-XI-1-2021, “BPVC Section XI-Rules for Inservice Inspection of Nuclear Power Plant Components, Division 1-Rules for Inspection and Testing of Components of Light Water-Cooled Plants,” (Reference 22-31) for Class 1, 2, and 3 components.

Environmental plant surveillance, leak detection, and leak testing will be identified as evidence of diversity for forewarning of structural failure. To periodically confirm that the material property values applied in design analysis remain appropriate throughout the plant lifetime, a programme of surveillance sampling will be specified. This provides samples for testing of mechanical properties, fracture toughness, and corrosion resistance properties to account for the effects of irradiation embrittlement and thermal ageing.

Further information relating to the surveillance of SSCs can be found in PSR Chapter 5 (Reference 22-8), PSR Chapter 6 (Reference 22-9), PSR Chapter 9A (Reference 22-10), and PSR Chapter 10 (Reference 22-11).

22.2.10 Operational Experience

OPEX provides a valuable source by which understanding of susceptibility to degradation is developed and maintained. The BWRX-300 benefits from this by including design enhancements considered for the ESBWR, as in IAEA SSG-30 (Reference 22-22). The materials selection reports include consideration of potential degradation mechanisms that may affect components of the BWRX-300 plant, as indicated by OPEX.

Some components of earlier BWR plants have experienced degradation by various mechanisms, notably by SCC. Enhancements that have been included to minimise the potential for SCC include selecting materials resistant to corrosion and optimisation of the manufacturing processes. The BWRX-300 operates with a water chemistry regime intended to prevent degradation. Specification of reactor coolant water chemistry is discussed in PSR Chapter 23 (Reference 22-18). In PSR Chapter 23 (Reference 22-18), Hydrogen Water Chemistry, On-Line NobleChem™ and Zinc Injection are the reference reactor chemistry regime for the UK BWRX-300.

The measures described in 006N5956 (Reference 22-28) are considered to effectively minimise the potential for SCC; however, it is acknowledged that the potential for such degradation cannot be eliminated for a 60-year period. For this reason, ISI forms an important element for control of degradation through periodic monitoring, particularly at locations where OPEX indicates vulnerability.

22.2.11 Safety Class 1 Components

For SC1 components, assurance of integrity is provided through compliance with the appropriate design codes and standards and also accounting for relevant OPEX from BWRs.

Substantiation of the structural reliability of SC1 components is based on demonstrating high quality of design and manufacture by compliance with relevant aspects of the ASME BPVC.

For SC1 components, the process of structural integrity classification establishes that protection against failure exists and that the potential consequences of failure are of a limited extent. The safety argument for SC1 components, therefore, concentrates on the effective prevention of failure and are founded on compliance with relevant requirements of the ASME BPVC, which provides suitably robust assurance of structural integrity. These cover quality design and manufacture, design code assessment, hydrostatic testing, and ISI.

The following sections of the ASME BPVC establish requirements that address key aspects of the BWRX-300 design:

- Section II Materials - ASME BPVC-II-2021, “Section II - Materials,” (Reference 22-32)
- Section III Rules for construction of nuclear facility components - ASME BPVC-III-NB-2021, “Section III - Rules for Construction of Nuclear Facility Components, Subsection NB-Class 1 Components," (Reference 22-33)
- Section V Non-destructive Examination - ASME BPVC-V-2021, “Section V -Non-destructive Examination,” (Reference 22-34)
- Section IX Welding, brazing, and fusing qualification - ASME BPVC-IX-2021, “Section IX - Welding, Brazing, and Fusing Qualifications,” (Reference 22-35)
- Section XI Rules for in-service inspection of nuclear reactor facility components - ASME BPVC-XI-1-2021 (Reference 22-31)

The ASME BPVC prescribes diverse measures to control the quality of design and manufacture and embodies extensive OPEX that is relevant to the BWRX-300 components. This ensures a structurally robust design and provides effective measures to prevent failure and to minimise, monitor, and control degradation by good design. Compliance with the ASME BPVC is therefore judged to provide a suitable means for assuring that the structural integrity of the BWRX-300 SC1 components can be maintained for the design lifetime.

Materials are specified and examined to effectively resist fracture and degradation. To demonstrate good choice of materials, evidence will be provided regarding their specification and procurement in accordance with the requirements of Section II of the ASME BPVC (Reference 22-32). This is intended to ensure that well proven materials are chosen that are resistant to fracture and of suitable composition to effectively limit the effect of through-life degradation. The material specification requirements include limitations on manufacturing techniques, the use of weld repairs, heat treatment, chemical composition, mechanical testing, inspection, and QA. Consideration of international OPEX and best practice will also be used to inform decisions on material selection and processing.

Section III of the ASME BPVC (Reference 22-33) includes a requirement to conduct structural analyses to support the design for a range of conditions. These include pressure, temperature, and mechanical loadings due to normal operating and test conditions, anticipated transients, and postulated accident conditions that could occur during operation. The evaluation of the service and testing conditions includes an evaluation of fatigue due to cyclic stresses. The results of these analyses will be identified as evidence to deterministically justify the structural integrity of BWRX-300 components against stress and fatigue limits established in the ASME BPVC, and thus confirm robust design.

Controls will be applied to ensure compliance with the design specification in manufacture, installation, and commissioning. The ASME BPVC includes measures to control quality of manufacture and installation. Relevant evidence will be provided in GDA Step 3 to include controls to ensure compliance with the welding procedures, testing of weld materials, and welder qualification with rules prescribed in Section III (Reference 22-33) and Section IX (Reference 22-35) of the ASME BPVC. End of manufacturing inspections will be performed to confirm the quality of the manufacturing.

Mechanical components and equipment that are classified as SC1 shall be provided with accessible openings for preservice inspection, ISI, and system pressure test, which support evaluations that justify the operational readiness of components and equipment as set forth within ASME BPVC-III-NB-2021 (Reference 22-33), ASME BPVC-XI-1-2021 (Reference 22-31), and IAEA SSG-74, “Maintenance, Testing, Surveillance and Inspection in Nuclear Power Plants,” (Reference 22-36).

Prior to the component or equipment leaving the manufacturer’s facility, mechanical components and equipment which require inspections and testing to satisfy ASME Section XI (Reference 22-31) requirements shall be examined by appropriate inspection methods as provided within ASME BPVC-III-1 (Reference 22-33), ASME BPVC-XI-1(Reference 22-31), ASME OM Code, “Operation and Maintenance of Nuclear Power Plants,” (Reference 22-37), IAEA SSG-74 (Reference 22-36), and NUREG-0800 Standard Review Plan 5.2.4.

In addition to a programme of inspection, the components will, where appropriate, be subjected to a hydrostatic over-pressurisation test and to a system hydrostatic test before entering service. The purpose of these tests is to confirm that the ability to sustain design pressure has not been compromised during manufacture and installation and that the design adequately prevents leakage. Hydrostatic tests will be specified in accordance with ASME BPVC Section III (Subsections NB and NC) (Reference 22-33).

In accordance with 006N6279 (Reference 22-30), ISI and monitoring are specified to effectively reveal degradation in good time. Timely forewarning of failure of SC1 components is provided by establishing ISI in accordance with the requirements Section XI of the ASME BPVC (Reference 22-31). Arrangements for leak monitoring, leak detection, and environmental monitoring will be identified, providing diverse means to reveal degradation and prompt corrective action.

22.2.12 Safety Class 2 and 3 Components

The process and outcome of classification of SC2 and 3 components is described in PSR Chapter 3 (Reference 22-7). The same claims presented in Appendix A are made for these components and are structured according to:

- Sound design and design code assessment to provide assurance of integrity
- High quality manufacture to ensure integrity is maintained throughout service
- Functional testing to confirm integrity at start of life
- ISI and monitoring to forewarn of failure

Section 3.6 of PSR Chapter 3 (Reference 22-7) provides the general design aspects used for Safety Class (SC) and Non-Safety Class (SCN) mechanical systems and components. It includes special considerations for mechanical components, dynamic testing, and analysis of SSCs, required codes for ASME BPVC, Section III, Division 1, Class 1, 2, and 3 components, Subsection NF for component supports, and Subsection NG for core support structures.

The requirements stated in these standards for design, manufacture, and inspection are used to underpin the structural integrity of these components.

22.3 Loading Conditions

Plant events affect mechanical systems and components. The load conditions due to the plant events and these load combinations are considered to evaluate the structural integrity.

The design load and loading combinations for mechanical systems and components are described in Section 3.6 of Attachment 1 to PSR Chapter 3 (Reference 22-7).

Service levels are classified into service conditions as indicated below based on frequency of occurrence and plant state:

- Normal: Planned Operation
- Upset: Anticipated Operational Occurrence (AOO)
- Emergency: Design Basis Accident (DBA)
- Faulted: Design Extension Condition (DEC)

The BWRX-300 utilises the four service levels used in ASME BPVC Section III (Reference 22-33), Division 1 & 2, Subsection NCA, Levels A, B, C, and D, as well as testing conditions in the design of fixed equipment. The design basis specifies the capabilities that are necessary for the plant in various operational states.

22.4 Summary of ALARP Justification

This section presents a high-level overview of how the ALARP principle has been applied for the structural integrity of metal components for a 60-year period of operation.

PSR Chapter 27 (Reference 22-19) presents a high-level approach taken for demonstrating ALARP across all aspects of the design and operation. It presents an overview of how the BWRX-300 design has evolved, the further options that have been considered across all technical areas resulting in design changes, and how these contribute to the overall ALARP case.

It is important to note that nuclear safety risks cannot be demonstrated to have been reduced ALARP within the scope of a Step 2 GDA. It is considered that the most that can realistically be achieved is to provide a reasoned justification that the BWRX-300 design aspects will effectively contribute to the development of a future ALARP statement. In this respect, this chapter contributes to the overall future ALARP case by demonstrating that:

- The chapter-specific arguments derived may be supported by existing and future planned evidence sources covering the following topics:
  - RGP has demonstrably been followed
  - OPEX has been taken into account within the design process
  - All reasonably practicable options to reduce risk have been incorporated within the design
- It supports its applicable level 3 sub-claims, defined within NEDC-34140P, “BWRX-300 UK GDA Safety Case Development Strategy (SCDS),” (Reference 22-38)

Probabilistic safety aspects of the ALARP argument are addressed within PSR Chapter 15.6 (Reference 22-14).

In this chapter, a structural integrity classification process is presented, and on the basis of UK ABWR GDA experience, some SC1 components will be sub-divided into HI and SC1. The safety case for the BWRX-300 design of each of the HI components is based on the safety claims listed in Appendix A. This multi-legged approach, as explained in Section 22.2 of this chapter, is in line with UK RGP and based upon the approach advocated by the UK TAGSI for HI components.

NEDC-34272P (Reference 22-23) describes an evaluation against the principle of ALARP of the tolerability of risks associated with the structural integrity of an HI component.

It is noted that the structural integrity aspects of safety significant metallic components on the BWRX-300 design are well understood and use proven technology. Hence, a significant aspect of demonstrating the application of RGP in the design of the BWRX-300 is to generally adopt the reference design for the ESBWR with necessary modifications to enhance safety margins even further where it is reasonably practicable to do so.

The BWRX-300 leverages the USNRC approved ESBWR design, proven in-use materials, off-the-shelf components, and design pressures and temperatures within the range of the existing BWR design and experience base.

As for material selection, as in 006N5956 (Reference 22-28), the most significant nuclear safety risks associated specifically with the structural integrity of metal components in the BWR designs over the plant lifetime are as follows.

Significant material degradation due to:

- Flow Accelerated Corrosion (FAC) and Erosion Corrosion (EC)
- Stress Corrosion Cracking

- Irradiation Assisted Stress Corrosion Cracking (IASCC)
- Neutron Irradiation Embrittlement
- Fatigue
- Other mechanisms (including general corrosion and pitting)

A key element of this chapter is to show that risks of accidents from safety significant metal structural failures can be mitigated through the approaches described; this is especially important for HI components because the claim is that the failure of an HI component can be discounted. This discounting of HI component failures is equivalent to the international concept of ‘practical elimination.’ Consistent with IAEA SSR 2/1, the BWRX-300 design is such that fault sequences that could lead to an early or large radioactive release are practically eliminated. Analyses demonstrating practical elimination are described in Table 15.9-C-1 of PSR Chapter 15.9 (Reference 22-15). This includes a sudden mechanical failure of the RPV that eliminates the capability to hold and cool the core.

Materials for BWRX-300 metallic components are appropriately selected to reduce risks caused by material degradations. It is recognised that material selection affects not only the potential for material degradation, but must also consider the interactions with the chosen BWR water chemistry (see PSR Chapter 23 (Reference 22-18)), and the need to minimise operating doses (see PSR Chapter 12 (Reference 22-12)). For example, use of low Cobalt material reduces Cobalt corrosion products and minimises the operating doses (see PSR Chapter 12 (Reference 22-12)) and the decommissioning source term (see PSR Chapter 21 (Reference 22-5)).

22.5 Conclusion

This chapter describes how the structural integrity of metallic SSCs that are significant to safety are assured for the BWRX-300. The process commences with the establishment of the safety functions required of a particular structural component, following which a system of safety structural classification is applied to determine the measures warranted to provide suitably robust assurance of structural integrity. The method of safety structural classification is based on postulated structural failure of the component with the associated loss of its safety function(s), taking into account both the direct and indirect unmitigated consequences of the failure.

This chapter describes the methods to establish SFCs and structural integrity classification in Section 22.1. It also provides reference to a topic report for the RPV that provides details of the methods and processes used to support the development of evidence to substantiate the structural integrity claims for BWRX-300 components. The structure and content of these varies according to classification, as described in Section 22.1.1. The nature and extent of evidence necessary to justify structural reliability is summarised for all classes of component that are significant to nuclear safety. This is based on compliance with appropriate design codes and standards, with supplementary measures identified to provide additional evidence of both defect avoidance and defect tolerance for components with the highest safety significance. The approach to specify load conditions and their combination for input to assessments that will support the structural integrity safety case is described in Section 22.3.

A key element of this chapter is to show that risks of accidents from all safety significant metallic structural failures can be mitigated through the approaches described; this is especially important for HI components because the claim is that the failure of an HI component can be discounted. This discounting of all HI component failures is similar to the international concept of ‘practical elimination’. Consistent with IAEA SSR 2/1, the BWRX-300 design is such that fault sequences that could lead to an early or large radioactive release are practically eliminated. Analyses demonstrating practical elimination are described in Table 15.9-C-1 of PSR Chapter 15.9 (Reference 22-15). This includes a sudden mechanical failure of the RPV that eliminates the capability to hold and cool the core.

Table A-1 provides a high-level summary of the claims, arguments and evidence covering the multiplicity of independent deterministic evidence to demonstrate a high confidence of low frequency of failure required of BWRX-300 HI components. Concepts such as excellence of design, excellence of manufacture, and rigorous testing are the basis for other safety significant structural metallic components, such as HI and SC1. The structural integrity safety analysis of metal components in this chapter is considered to support a future demonstration that the risks of failure of metal structural components may be reduced ALARP.

Table 22-1: Structural Integrity Classification

Safety Class	Consequences of Failure
HI	Severe core damage and large off-site release of radiation.
SC1	Localised damage to fuel. Minor off-site release. Significant release within nuclear island.
SC2 & 3	No core damage. Fault within capability of protective systems. Contamination within nuclear island.

22.6 References

22-1 005N9751, "BWRX-300 General Description," Rev F, GE-Hitachi Nuclear Energy, Americas, LLC, December 2023.

22-2 GA91-9101-0101-08000, "UK ABWR Generic Design Assessment, Generic PCSR Chapter 8: Structural Integrity," Hitachi-GE, Rev C, 2017.

22-3 005N9461, "BWRX-300 Structures Systems, and Components Safety Classification," Rev 4, GE-Hitachi Nuclear Energy, Americas, LLC.

22-4 "The Demonstration of Incredibility of Failure in Structural Integrity Safety Cases," R Bullough, F M Burdekin, O V J Chapman, V R Green, D P G Lidbury, J N Swingler, R Wilson, International Journal of Pressure Vessels and Piping 78, pages 539-552, 2001.

22-5 NEDO-34193, "BWRX-300 UK GDA Chapter 21 – Decommissioning and End of Life Aspects," Rev B, GE-Hitachi Nuclear Energy, Americas, LLC.

22-6 IAEA SSG-61, "Format and Content of the Safety Analysis Report for Nuclear Power Plants," International Atomic Energy Agency, 2021.

22-7 NEDO-34165, "BWRX-300 UK GDA Chapter 3 – Safety Objectives and design Rules for SSCs," Rev B, GE-Hitachi Nuclear Energy, Americas, LLC.

22-8 NEDO-34167, "BWRX-300 UK GDA Chapter 5 – Reactor Coolant System and Associated Systems," Rev B, GE-Hitachi Nuclear Energy, Americas, LLC.

22-9 NEDO-34168, "BWRX-300 UK GDA Chapter 6 – Engineered Safety Features," Rev B, GE-Hitachi Nuclear Energy, Americas, LLC.

22-10 NEDO-34171, "BWRX-300 UK GDA Chapter 9A – Auxiliary Systems," Rev B, GE-Hitachi Nuclear Energy, Americas, LLC.

22-11 NEDO-34173, "BWRX-300 UK GDA Chapter 10 – Steam and Power Conversion Systems," Rev B, GE-Hitachi Nuclear Energy, Americas, LLC.

22-12 NEDO-34175, "BWRX-300 UK GDA Chapter 12 – Radiation Protection," Rev B, GE-Hitachi Nuclear Energy, Americas, LLC.

22-13 NEDO-34176, "BWRX-300 UK GDA Chapter 13 – Conduct of Operations," Rev B, GE-Hitachi Nuclear Energy, Americas, LLC.

22-14 NEDO-34184, "BWRX-300 UK GDA Chapter 15.6 – Probabilistic Safety Assessment," Rev B, GE-Hitachi Nuclear Energy, Americas, LLC.

22-15 NEDO-34187, "BWRX-300 UK GDA Chapter 15.9 – Summary of Results of the Safety Analyses," Rev B, GE-Hitachi Nuclear Energy, Americas, LLC.

22-16 NEDO-34188, "BWRX-300 UK GDA Chapter 16 – Operational Limits and Conditions of Safe Operation," Rev B, GE-Hitachi Nuclear Energy, Americas, LLC.

22-17 NEDO-34190, "BWRX-300 UK GDA Chapter 18 – Human Factors Engineering," Rev B, GE-Hitachi Nuclear Energy, Americas, LLC.

22-18 NEDO-34195, "BWRX-300 UK GDA Chapter 23 – Reactor Chemistry," Rev B, GE-Hitachi Nuclear Energy, Americas, LLC.

22-19 NEDO-34197, "BWRX-300 UK GDA Chapter 27 – ALARP Evaluation," Rev B, GE-Hitachi Nuclear Energy, Americas, LLC.

22-20 006N5064, "BWRX-300 Safety Strategy," Rev 6, GE-Hitachi Nuclear Energy, Americas, LLC.

22-21 IAEA SSR 2/1, "Safety of Nuclear Power Plants: Design," International Atomic Energy Agency, 2016.

22-22 IAEA SSG-30, "Safety Classification of Structures, Systems and Components in Nuclear Power Plants," International Atomic Energy Agency, 2014.

22-23 NEDC-34272P, "BWRX-300 UK GDA Topic Report – Reactor Pressure Vessel Structural Integrity Substantiation Methodology," Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

22-24 GA91-9201-0003-00054 (RD-GD-0001), "Structural Integrity Classification Procedure," Hitachi-GE, Rev 0, March 2014.

22-25 GA91-9201-0003-00057 (G-TY-53082), "Inspection Qualification Strategy," Hitachi-GE, Rev 0, March 2014.

22-26 "Assessment of the Integrity of Structures Containing Defects," EDF Energy, R6 Rev 4, 2023.

22-27 GA91-9201-0003-00056 (RD-GD-0003), "Defect Tolerance Assessment Plan," Hitachi-GE, Rev 1, June 2016.

22-28 006N5956, "Materials and Process Controls," Rev 1, GE-Hitachi Nuclear Energy, Americas, LLC.

22-29 006N7441, "BWRX-300 Reactor Pressure Vessel Fabrication Requirements," Rev 1, GE-Hitachi Nuclear Energy, Americas, LLC.

22-30 006N6279, "BWRX-300 In Service Inspection Requirements," Rev 1, GE-Hitachi Nuclear Energy, Americas, LLC.

22-31 ASME BPVC-XI-1-2021, "BPVC Section XI-Rules for Inservice Inspection of Nuclear Power Plant Components, Division 1-Rules for Inspection and Testing of Components of Light Water-Cooled Plants," American Society of Mechanical Engineers, 2021.

22-32 ASME BPVC-II-2021, "Section II-Materials," American Society of Mechanical Engineers, 2021.

22-33 ASME BPVC-III-NB-2021, "Section III-Rules for Construction of Nuclear Facility Components, Subsection NB-Class 1 Components," American Society of Mechanical Engineers, 2021.

22-34 ASME BPVC-V-2021, "Section V-Non-destructive Examination," American Society of Mechanical Engineers, 2021.

22-35 ASME BPVC-IX-2021, "Section IX-Welding, Brazing, and Fusing Qualifications," American Society of Mechanical Engineers, 2021.

22-36 IAEA SSG-74, "Maintenance, Testing, Surveillance and Inspection in Nuclear Power Plants," International Atomic Energy Agency, 2022.

22-37 ASME OM Code, "Operation and Maintenance of Nuclear Power Plants," American Society of Mechanical Engineers, 2022.

22-38 NEDC-34140P, "BWRX-300 UK GDA Safety Case Development Strategy," Rev 0, GE-Hitachi Nuclear Energy, Americas, LLC.

22-39 "Safety Assessment Principles (SAPs)", Office for Nuclear Regulation, 2014 edition (Rev 1, 2020).

22-40 NEDC-34137P, "BWRX-300 UK GDA BWRX-300 Design Evolution," Rev 0, GE-Hitachi Nuclear Energy, Americas, LLC.

22-41 006N3441, “BWRX-300 Applicable Codes, Standards, and Regulations List,” Rev 3, GE-Hitachi Nuclear Energy, Americas, LLC.

22-42 NEDC-34139P, “BWRX-300 UK GDA Step 1 Codes and Standards Report,” Rev 1, GE-Hitachi Nuclear Energy, Americas, LLC.

22-43 006N8706, “BWRX-300 Construction Strategy Report,” Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

22-44 006N3139, “BWRX-300 Design Plan,” Rev 5, GE-Hitachi Nuclear Energy, Americas, LLC.

22-45 NEDC-34274P, “BWRX-300 UK GDA Forward Action Plan,” Rev 2, GE-Hitachi Nuclear Energy, Americas, LLC.

APPENDIX A CLAIMS, ARGUMENTS AND EVIDENCE

Claims, Arguments and Evidence

The Office for Nuclear Regulation (ONR) "Safety Assessment Principles (SAPs)," (Reference 22-39) identify ONR's expectation that a safety case should clearly set out the trail from safety claims, through arguments to evidence. The Claims, Arguments and Evidence (CAE) approach can be explained as follows:

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

The GDA CAE structure is defined within NEDC-34140P (Reference 22-38) 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.

The Level 3 sub-claims that this chapter demonstrates compliance against are identified within NEDC-34140P (Reference 22-38) and are as follows:

> *2.1.2: The design of the system/structure has been substantiated to achieve the safety functions in all relevant operating modes.*
>
> *2.1.3: The system/structure design has been undertaken in accordance with relevant design codes and standards (RGP) and design safety principles, and taking account of Operating Experience to support reducing risks ALARP*
>
> *2.1.4: System/structure performance will be validated by suitable testing throughout manufacturing, construction, and commissioning.*
>
> *2.1.5: Ageing and degradation mechanisms will be identified and assessed in the design. Suitable examination, inspection, maintenance, and testing will be specified to maintain systems/structures fit-for-purpose through-life.*
>
> *2.1.6: The BWRX will be designed so that it can be decommissioned safely, using current available technologies, and with minimal impact on the environment and people*
>
> *2.4.1: Relevant Good Practice (RGP) has been taken into account across all disciplines*
>
> *2.4.2: Operational Experience (OPEX) and Learning from Experience (LfE) has been taken into account across all disciplines*
>
> *2.4.3: Optioneering (all reasonably practicable measures have been implemented to reduce risk).*

In order to facilitate compliance demonstration against the above Level 3 sub-claims, this PSR chapter has derived a suite of arguments that comprehensively explain how their applicable Level 3 sub-claims are met (see Table A-1 below).

It is not the intention to generate a comprehensive suite of evidence to support the derived arguments, as this is beyond the scope of GDA Step 2. However, where evidence sources are available, examples are provided.

Table A-1: Structural Integrity Claims and Arguments

<table>
<tr><th>L3 No.</th><th>Level 3 Chapter Claim:</th><th>Chapter 22 Arguments:</th><th>PSR Chapters & Subsections Where the Arguments are Supported:</th></tr>
<tr><td colspan="4">2.1: The functions of systems and structures have been derived and substantiated taking into account RGP and OPEX, and processes are in place to maintain these through-life. (Engineering Analysis)</td></tr>
<tr><td rowspan="4">2.1.2</td><td rowspan="4">The design of the system/structure has been substantiated to achieve the safety functions in all relevant operating modes.</td><td>Safety functions associated with the relevant SSC have been substantiated during normal operating conditions (including design codes and standards compliance).</td><td>Safety functions are identified in PSR Chapter 3 (Reference 22-7) and PSR Chapter 15.6 (Reference 22-14).
Reactor Coolant System and Associated Systems (PSR Chapter 5 (Reference 22-8)).
Engineered Safety Features (PSR Chapter 6 (Reference 22-9)).
Auxiliary Systems (PSR Chapter 9A (Reference 22-10)).
Steam and Power Conversion Systems (PSR Chapter 10 (Reference 22-11)).</td></tr>
<tr><td>A record of safe BWR plant operation and continuous improvement demonstrates a well-founded design.</td><td>NEDC-34137P, “BWRX-300 UK GDA BWRX-300 Design Evolution,” (Reference 22-40).</td></tr>
<tr><td>Safety functions associated with the relevant SSC have been substantiated during hazard and fault conditions.</td><td>Safety functions are identified in PSR Chapters 3 and 15.
Means of substantiation will be included in PSR Chapters 5, 6, 9A, and 10.</td></tr>
<tr><td>Any shortfalls in safety function substantiation have been identified and assessed to identify any reasonably practicable means to reduce risk.</td><td>This argument is out of the scope of GDA Step 2 and will be addressed during a site-specific stage (when evidence is developed).</td></tr>
<tr><td>2.1.3</td><td>The system/structure design has been undertaken in accordance with relevant design codes and</td><td>Design evolutions of SSCs have been considered including relevant BWR OPEX, and any reasonably practicable changes to reduce risk have been implemented.</td><td>NEDC-34137P (Reference 22-40).
See PSR Chapters 5, 6, 9A, and 10.</td></tr>
</table>

L3 No.	Level 3 Chapter Claim:	Chapter 22 Arguments:	PSR Chapters & Subsections Where the Arguments are Supported:
	standards (RGP) and design safety principles and taking account of Operating Experience to support reducing risks ALARP.	The SSCs have been designed in accordance with relevant codes and standards (RGP).	006N3441, “BWRX-300 Applicable Codes, Standards, and Regulations List,” (Reference 22-41). NEDC-34139P, “BWRX-300 UK GDA Step 1 Codes and Standards Report,” (Reference 22-42). See PSR Chapters 5, 6, 9A, and 10.
		The SSCs have been designed in accordance with an appropriate suite of design safety principles.	The GEH Safety and Design Principles are documented in the BWRX-300 Safety Strategy, supplemented by the BWRX-300 General Description. These principles are also presented within PSR Chapter 3 (Reference 22-7). 006N5064 (Reference 22-20). 005N9751 (Reference 22-1).
		For HI components, the integration of defect tolerance assessment, NDT, high reliability, and lower bound properties support the avoidance of fracture demonstration.	See NEDC-34272P (Reference 22-23).
		Calculated stresses are conservatively set based on a comprehensively specified load schedule.	This argument is out of GDA Step 2 scope and will be addressed during the site-specific stage.
		The design is to tolerate the specified environmental parameters.	See Section 22.2.5.
		The specified operating limits are conservatively taken into account within the design as determined by well-established design methods.	See PSR Chapter 3 (Reference 22-7), Attachment 1, Section 3.6 for design transients.

L3 No.	Level 3 Chapter Claim:	Chapter 22 Arguments:	PSR Chapters & Subsections Where the Arguments are Supported:
		Components are designed for ease of inspection.	Mechanical components and equipment, including containment penetrations, heat exchangers, pipe supports, pumps, valves, and vessels, which are classified as Safety Class 1 and ASME BPVC-III-NB-2021 and ASME BPVC-III-NCD-2021 shall be provided with accessible openings for preservice inspection, ISI and system pressure test, which support evaluations that justify the operational readiness of components and equipment, and in GA91-9201-0003-00057 (G-TY-53082) (Reference 22-25).
		Effective and proven processes are specified to achieve high quality of manufacture.	See Section 22.2.6.
2.1.4	System/structure performance will be validated by suitable testing throughout manufacturing, construction, and commissioning.	SSC pre-commissioning tests (e.g., NDT) validate the relevant performance requirements.	See PSR Chapters 3, 5, 6, 9A, and 10.
		SSC commissioning tests (e.g., system level pressure and leak tests) validate the relevant performance requirements.	ASME V Pre-Service Inspection: Examination Categories & Methods (5.11.4). See PSR Chapters 5, 6, 9A, and 10. Note: This is considered to be beyond the scope of GDA Step 2 to define.
		SSCs are manufactured, constructed, and commissioned in accordance with QA arrangements appropriate to their safety classification.	PSR Chapter 3 (Reference 22-7) defines this approach. Quality Assurance: 5.3.1 Nuclear Boiler System Configuration; 5.3.5 Testing and Maintenance; 5.10.2 Main Steam Containment Isolation Valves. 006N8706, “BWRX-300 Construction Strategy Report,” (Reference 22-43) describes the high-level construction QA and quality control arrangements and responsibilities.
		High reliability manufacturing will be qualified by NDT to provide assurance of no structural defects of concern.	See NEDC-34272P (Reference 22-23). NDE methods shall be accordance with ASME BPVC-V-2021 (Reference 22-34), ASME BPVC-XI-1-2021 (Reference 22-31).

L3 No.	Level 3 Chapter Claim:	Chapter 22 Arguments:	PSR Chapters & Subsections Where the Arguments are Supported:
		Components are manufactured through judicious material selection.	See 006N5956 (Reference 22-28).
2.1.5	Ageing and degradation mechanisms will be identified and assessed in the design. Suitable examination, inspection, maintenance, and testing will be specified to maintain systems/structures fit-for-purpose through-life.	SSC ageing and degradation mechanisms will be identified during SSC design. These will be assessed to determine how they could potentially lead to SSC failure.	PSR Chapter 3 sub-sections 3.1.13, 3.4.4.1, 3.9.1, 3.9.2, and 3.9.3 and Chapter 13 sub-section 13.3.8. OPEX on BWRX-300: 5.2.1 Basis of Material Selection and Component Fabrication; 5.2.3 Overview of Reactor Pressure Vessel and Pressure Boundary Components; 5.2.5 Overview of Core Structural Component Materials; 5.2.7 Gasket, Seal, and Fastener Materials; 5.9.4 Reactor Pressure Vessel Internal Supports. Mitigation against degradation of RPV and Pressure Boundary Components: 5.2.4 Radiation embrittlement, general corrosion, Flow Accelerated Corrosion (FAC). 5.2 Material selection. 5.12.1 Hydrogen water chemistry (reduce the risk of IGSCC in reactor vessel internals). 5.2.6 Welding process improvements increase margins against SCC.
		Appropriate Examination, Maintenance, Inspection and Testing (EMIT) arrangements will be specified taking into account SSC ageing and degradation mechanisms.	This is considered to be out of the scope of a 2-Step GDA, where the design maturing is at a concept stage. However, early project examples of such considerations are included within the following reports: ASME XI Inservice Inspection: 5.3.4 RCS System Operations; 5.7.7 Monitoring, Inspection, Testing, and Maintenance. ASME Operation & Maintenance: 5.3.5 RCS Testing and Maintenance. 006N6279 (Reference 22-30).
		The SSCs that cannot be replaced have been shown to have adequate life.	PSR chapters 3 and 13 describe BWRX-300's ageing management arrangements for systems. Specifically, PSR Chapter 3 sub-sections 3.1.13, 3.4.4.1, 3.9.1, 3.9.2, and 3.9.3 apply, along with Chapter 13 sub-section 13.3.8.
		Ageing and degradation OPEX will be considered as part of the design stage component/materials selection	OPEX on BWRX-300: 5.2.1 Basis of Material Selection and Component Fabrication; 5.2.3 Overview of Reactor Pressure Vessel and Pressure Boundary Components; 5.2.5 Overview of Core Structural Component Materials; 5.2.7

L3 No.	Level 3 Chapter Claim:	Chapter 22 Arguments:	PSR Chapters & Subsections Where the Arguments are Supported:
		process in order to mitigate SSC failure risk.	Gasket, Seal, and Fastener Materials; 5.9.4 Reactor Pressure Vessel Internal Supports. NEDC-34137P (Reference 22-40). PSR chapters 3 and 13 describe BWRX-300's ageing management arrangements for systems. Specifically, PSR Chapter 3 sub-sections 3.1.13, 3.4.4.1, 3.9.1, 3.9.2, and 3.9.3 apply, along with Chapter 13 sub-section 13.3.8.
		Provision is made for monitoring during operation to forewarn of failure.	See PSR Chapters 5, 6, 9A, and 10
2.1.6	The BWRX will be designed so that it can be decommissioned safely, using current available technologies, and with minimal impact on the environment and people.	SSC decommissioning is considered at the design stage to ensure that safe decommissioning may take place.	OPEX demonstrates that decommissioning of reactor facilities is facilitated if considered during the design phase: [1] Materials are selected to minimise the quantities of radioactive waste and assisting decontamination. [2] Plant layout is designed to facilitate access for decommissioning or dismantling activities. [3] Future potential requirements for storage of radioactive waste. See PSR Chapter 21 (Reference 22-5).
		SSCs are designed in order to minimise impacts on people and the environment during decommissioning.	BWRX-300 Decommissioning and End of Life Aspects (PSR Chapter 21 (Reference 22-5)). Materials selection to reduce in-service activation is an example of how decommissioning dose uptakes could be minimised, see 006N5956 (Reference 22-28).
2.4 Safety risks have been reduced as low as reasonably practicable			
2.4.1	RGP has been taken into account across all disciplines.	Relevant SSC codes and standards (RGP) are identified.	006N3441 (Reference 22-41). NEDC-34139P (Reference 22-42).
		SSCs have been designed in accordance with relevant codes and standards (RGP).	006N3441 (Reference 22-41). NEDC-34139P (Reference 22-42). Also, this PSR chapter and PSR Chapter 3 discuss the codes and standards to which the SCCs have been designed.

L3 No.	Level 3 Chapter Claim:	Chapter 22 Arguments:	PSR Chapters & Subsections Where the Arguments are Supported:
		Any shortfalls in codes and standards compliance are identified and assessed to reduce risks ALARP.	Out of the scope of this PSR chapter for GDA Step 2.
2.4.2	Operational Experience (OPEX) and Learning from Experience (LfE) has been taken into account across all disciplines.	Design improvements to SSCs have been identified considering relevant OPEX and LfE.	006N3139, "BWRX-300 Design Plan," (Reference 22-44), section 4.7, mentions the GEH OPEX process CP-16-101. The design process incorporates applicable OPEX to mitigate nuclear design and construction risk, in accordance with CP-16-101 and the BWRX-300 OPEX/lessons learned programme. Operating experience sources include INPO, EPRI and BWROG. Construction experience and improved construction methods from previous large projects are also used to improve the quality and efficiency of the construction effort. NEDC-34137P (Reference 22-40).
		Any reasonably practicable design changes to reduce risk have been implemented.	The BWRX-300 has benefited from decades of OPEX in developing its design. NEDC-34137P (Reference 22-40).
2.4.3	Optioneering (all reasonably practicable measures have been implemented to reduce risk).	Design optioneering has been performed in accordance with an approved process.	006N3139 (Reference 22-44).
		Design optioneering has considered all reasonably practicable measures.	006N3139 (Reference 22-44). NEDC-34137P (Reference 22-40).
		Any reasonably practicable design changes to reduce risk have been implemented.	NEDC-34137P (Reference 22-40).

APPENDIX B FORWARD ACTION PLAN

The Forward Action Plan (FAP) is not required to capture the ‘normal business of Safety, Security, Safeguards, and Environmental case development as the design progresses from concept to design for construction and commissioning. FAP items can arise from several sources:

- Assumptions and commitments made in the GDA submissions that will require future verification/implementation, for example, by the future constructor and/or plant operator
- A gap in the underpinning of the GDA submissions currently under development
- A potential gap in a future phase of submissions if additional work is not performed
- A gap identified by the regulators and communicated to the Requesting Party through a Regulatory Query or Regulatory Observation

The FAP item in Table B-1 below originates from Chapter 15.7 and is also included within the project’s FAP report, NEDC-34274P (Reference 22-45).

Table B-1: Structural Integrity Forward Action Plan Items

FAP No.	Finding	Forward Action Plan Item	Delivery Phase
PSR15.7-63	The application of a Break Exclusion Zone as described in Chapter 3 Attachment 1 Section 3.2.4.1 does not align to UK RGP. To align with UK RGP, the presentation of justification / substantiation should consider: - Looking at the developments in the standard plant, and the supporting justification / substantiation. - Look to represent the information in a Claims, Arguments, Evidence structure, developing muti-legged arguments for the items where the highest integrity claims are made. - Look to incorporate the information presented in UK PSR Chapter 22 into the multi-legged arguments and identification of potential sources of future evidence. - Review the proposed approach to consider the challenges from other disciplines (e.g. radiological protection) to inform what is reasonably practicable. - Review UK RGP to determine if any further potential gaps arise and understand how or when these can be addressed. - Ensure the information can demonstrate the approach is ALARP.	The action should look to develop an approach for the highest integrity components in the assessment of internal hazards, this has been addressed for GDA Step 2 via the gap recorded in PSR15-3, for which NEDC-34357P, “BWRX-300 UK GDA Safety Case Manual Specification,” Rev 0, has been produced. Following that the action should look to develop detailed methodologies for PCSR/Pre-Construction Environmental Report (PCER) that incorporates the considerations listed in the Finding in order to establish what further work might be needed to address any gaps or shortfalls. This would benefit from input from the appropriate GE Hitachi Nuclear Energy (GEH) piping and structural integrity teams.	For PCSR/PCER