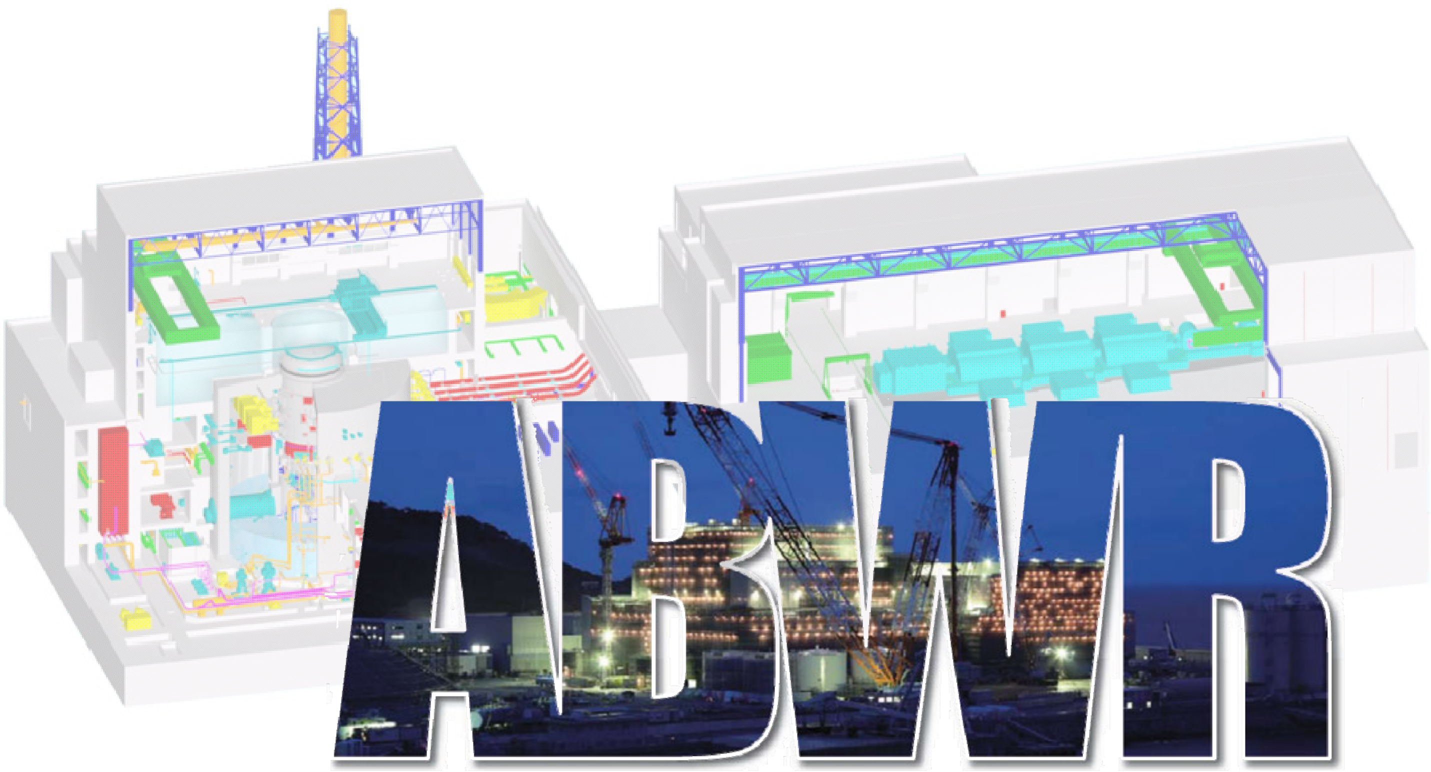
Form10/00

UK ABWR

Document ID	:	GA91-9101-0101-08000
Document Number	:	RE-GD-2043
Revision Number	:	A

UK ABWR Generic Design Assessment

Generic PCSR Chapter 8 : Structural Integrity



Hitachi-GE Nuclear Energy, Ltd.

DISCLAIMERS

Proprietary Information

Copyright

Table of Contents

8.1 Introduction

This chapter describes the approach for the demonstration of the structural integrity of metal systems, structures and components (SSCs) of the United Kingdom (UK) Advanced Boiling Water Reactor (ABWR). It describes the structural integrity classification of SSCs, provides a summary of the claims, argument and evidence that make up the safety case, describes the role of codes and standards according to their safety class and the approach for the combination of loads.

The structural integrity of metal SSCs has to be demonstrated. The genesis of the ABWR design is shown in [Ref-1] and summaries of the design of significant ABWR components are shown in [Ref-2], [Ref-3], [Ref-4] and [Ref-5]. Evidence of sound engineering practice is identified to establish that it is of sufficient quality and diversity to assure structural integrity through good design, manufacture and throughout operation to the end of plant life. This is intended to support the Generic Design Assessment (GDA) of the ABWR in UK. The GDA progresses in a series of steps and the current revision of this chapter is submitted at Step 2 of GDA, reflecting the current state of development of the UK ABWR design. Future revisions will provide increased detail as design substantiation matures throughout the GDA.

[Ref-6] describes the overall approach for the demonstration of structural integrity. This is founded on the clear understanding of the potential radiological consequences of any postulated failure mode. Accordingly, a structural integrity classification methodology has been developed. This classification scheme provides a structured and systematic basis for establishing the level of rigour applied during the design assessment, material procurement, fabrication, in-manufacture inspection, testing and in-service testing, maintenance and inspection of the component, and the development of the safety case arguments. This classification scheme will be applied in a systematic manner across the metallic components both within the nuclear island and across the Balance of Plant (BoP).

For a component with a high degree of redundancy or diverse means of protection, where the radiological consequences of failure are minimal, Hitachi-GE proposes that a structural reliability claim based on the demonstration of good design, and manufacture in accordance with recognised nuclear design standards will be appropriate. However, where limited or no protection can be provided, i.e. where it is impracticable to adequately mitigate the consequences of failure, it is necessary for a higher degree of reliability to be sought and substantiated. For the UK ABWR, this will require measures to be implemented over and above the design code requirements, with a particular emphasis on demonstration that a structure is as defect free as possible and is defect tolerant. For those components where the claim needs to be made that the likelihood of gross failure is so low that it can be discounted, the highest levels of structural reliability will be substantiated.

8.2 Scope

The scope is to consider all SSCs that are important to nuclear safety, as identified by classification, with respect to their structural integrity under all credible normal, fault and accident conditions. This will address the avoidance of defects, by good design and quality of manufacture, effective control of potential degradation mechanisms and, where appropriate, contingency measures against unexpected conditions or through-life degradation. The scope of work to substantiate structural integrity claims is summarised as follows:

- Identify safety functional requirements (SFRs) and categorise these according to their importance to safety.
- Identify SSCs that deliver each safety function and classify these according to the importance of their SFRs.
- Establish suitably rigorous requirements for design, construction, and operation, according to classification.

8.3 Objectives

The objective is to justify the structural integrity of the components included within the scope for a 60-year period of operation, and demonstrate that plant risk remains both tolerable and as low as reasonably practicable for the design lifetime. This will be substantiated by satisfying the safety design bases for each SSC, according to the importance of the SFRs they provide to safety.

8.4 Safety Functional Requirements

The SFRs of any SSC determine how nuclear safety will be maintained under all design basis conditions. The SFRs will be identified for the respective SSCs during Step 3 of GDA. The high level safety claims, performance and safety design bases for the reactor coolant system are discussed in Chapter 12 of this Pre-Construction Safety Report (PCSR), and will provide the basis for subsequently identifying the specific level SFRs for the SSCs.

For structural integrity components, SFRs will be identified by a review of the performance and safety design bases relevant to a particular component to establish where, and to what extent, structural integrity must be justified to maintain the required safety functional performance. A future update to this PCSR will tabulate the SFRs allocated to each plant component. The range of operating conditions included within the design basis for each component will include all conditions associated with normal operational, test and fault conditions.

8.5 Structural Integrity Classification

The structural reliability of UK ABWR SSCs will be justified according to the consequences of their failure, as established by a system of component structural integrity classification. Sub-chapter 5.4 describes the safety classification scheme for UK ABWR SSCs and establishes three classes based on safety significance.

The frequency and consequences of failure of Class 1 components will vary significantly. As the risk of failure varies so will the required assurance of structural integrity. In order to identify where the very highest standards of structural integrity should apply, [Ref-7] describes a refined scheme of classification to be adopted which sub-divides Class 1 into three classes; Very High Integrity (VHI), High Integrity (HI) and Standard Class 1.

Table 8.5-1 illustrates the criteria, in terms of consequences of failure and availability of protection that will determine classification. VHI is assigned to failure modes for which there is no protection, where failure is intolerable and where it is not reasonably practicable to provide protection. 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.

Structural Integrity Classification Process

The process identified in [Ref-8] is based on consideration of the components safety function, postulated failure mode, direct consequences of failure, lines of protection and the indirect consequences of failure, such as those arising from the generation of energetic missiles or pipe whip. [Ref-7] describes an approach for considering each of these, a methodology for recording the data and a process of expert review. The process of classification will be carried out in several stages, beginning in Step 2 of GDA, with the outcome to be reported in a future revision of this Sub-chapter.

Table 8.5-1: Structural Integrity Classes

Class	Protection	Consequence
VHI	None	Severe core damage and large off-site release of radiation
HI	1 line, no redundancy	Severe core damage. Containment protects against large off-site release. Limited release of radioactive material.
SC 1	1 line, redundancy	Localised damage to fuel. Minor off-site release. Significant release within nuclear island.
SC2 & 3	2 lines with diversity	No core damage. Fault within capability of protective systems. Contamination within nuclear island.

8.6 Component Safety Reports

This Sub-chapter summarises the approach which will be taken to demonstrate suitably robust structural integrity for Class 1, 2, 3, HI, and VHI SSCs of the UK ABWR. This will be supplemented in a later step of GDA through revision of this chapter to include a number of appendices. The standards by which structural integrity is assured will reflect the functional reliability requirements of the SSCs, commensurate with their safety classification. The structural integrity of all Class 1, HI and VHI SSCs will, as a minimum, be justified by evidence of compliance with the requirements of a well established and appropriate design code. Supplementary evidence to support more exacting integrity claims will be provided for HI and VHI components. The structure of the component safety reports, and the approach for each class, is described below.

Structure of Component Safety Reports

Several component safety reports will be appended to a future revision of this chapter, these documents will be presented as topic reports. Individual topic reports will be produced to provide detailed justification of the structural integrity for each HI and VHI component. Since the process of classification is not yet complete, the list of VHI and HI components has not been established and will be identified in a future revision of this sub-chapter.

The structure of the topic reports for VHI and HI components will be consistent with that recommended by the UK Technical Advisory Group on Structural Integrity (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 and VHI components will be structured according to five distinct claims as follows:

Claim 1: Structural Integrity is assured by good design

Claim 2: Structural Integrity is assured by material selection and quality manufacturing

Claim 3: Functional testing provides a demonstration of integrity at start of life

Claim 4: Through life integrity is demonstrated by analysis and inspection

Claim 5: Inspection and Monitoring regularly validate integrity through life

Each claim will be supported by a series of arguments, which will each be substantiated by identification of robust and diverse evidence, typically compiled as a dossier of technical information, data and analyses reports. Since the precise nature and quantity of the safety arguments to be presented in support of each claim will vary between components, a more detailed structure is not provided at this step. A summary of the evidence that will typically be identified in support of each claim for HI and VHI components is provided later in this sub-chapter. It is planned that the first topic report to be produced will address the structural integrity of the reactor pressure vessel; a first revision of this topic report will be produced in GDA Step 3 to establish the claims-arguments-evidence structure applied in the case of this component.

Structural integrity of Class 1 components will be founded on demonstration of compliance with the requirements of an established nuclear design code. It is planned that a single topic report, or a small number of reports, will be developed to identify the codes and standards applied to the design and manufacture of Class 1 components, and identify evidence of compliance with the applicable sections of these codes and standards. A summary of the evidence that will typically be identified to demonstrate design code compliance for Class 1 components is provided later in this sub-chapter.

A similar approach will be adopted for Class 2 and 3 components within the BoP, and a single topic report is planned to address structural integrity of BoP components.

Class 1 Components

Substantiation of the structural reliability of the Class 1 components will be based on demonstrating high quality of design and manufacture by compliance with relevant aspects of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (the ASME B&PV Code). The precise edition, addenda and code cases applied will be specified in the appendices to be added in a later step of GDA.

The ASME B&PV Code prescribes rules governing the design, fabrication, and inspection of boilers and pressure vessels that are well established for application in the nuclear industry. The following sections of the ASME B&PV Code establish requirements that address key aspects of ABWR design:

Section II	Materials
Section III	Rules for construction of nuclear components
Section V	Non-destructive examination
Section IX	Welding and brazing qualification
Section XI	Rules for in-service inspection of nuclear components

For Class 1 components, the process of structural integrity classification will establish that protection against failure exists and that the potential consequences of failure are of a limited extent. The safety argument for Class 1 components will therefore concentrate on the effective prevention of failure. Four broad claims will be used to structure the topic reports for Class 1 components, each founded on compliance with relevant requirements of the ASME B&PV Code which provides suitably robust assurance of structural integrity. These will cover quality design and manufacture, design code assessment, hydrostatic testing and in service inspection.

The ASME B&PV Code prescribes diverse measures to control quality of design and manufacture and embodies extensive operating experience that is relevant to the ABWR 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 B&PV Code is therefore judged to provide a suitable means for assuring that the structural integrity of the ABWR Class 1 components will be maintained for the design lifetime.

Materials will be 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 B&PV Code. This is intended to ensure that well proven materials are chosen which 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 quality assurance. Evidence will also be provided to demonstrate compliance with the mechanical testing requirements of Section III of the ASME B&PV Code.

Section III of the ASME B&PV Code 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 ABWR components against stress and fatigue limits established in the ASME B&PV Code and confirm robust design.

Controls will be applied to ensure compliance with the design specification in manufacture, installation and commissioning. The ASME B&PV Code includes measures to control quality of manufacture and installation. Relevant evidence will include controls to ensure compliance with the welding procedures, testing of weld materials and welder qualification with rules prescribed in Section III and Section IX of the code.

Pre-service inspection and testing will be specified to confirm quality of design and manufacture and these will be applied to confirm robust structural integrity before the component enters service. Section III of the ASME B&PV Code includes control of manufacturing inspections to confirm the absence of defects, and functional testing to confirm pressure boundary integrity.

In addition to a programme of inspection the Class 1 components will, where appropriate, be subjected to a hydrostatic overpressurisation 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 Section III (Subsections NB and NC) of the ASME B&PV Code.

In-service inspection (ISI) and monitoring will be specified to effectively reveal degradation in good time. Evidence of ISI that will effectively provide timely forewarning of failure for Class 1 components will be provided by establishing that ISI will be specified in accordance with the requirements of the ASME B&PV Code Section XI. Arrangements for leak monitoring, leak detection and environmental monitoring will be identified, providing diverse means to reveal degradation and prompt corrective action.

HI Components

The failure of HI components can lead to severe radiological consequences, but the process of structural integrity classification will identify evidence that effective containment exists to limit the off site consequences. It is necessary that the structural integrity of the HI regions is substantiated to a higher degree of rigour than that required for the Standard Class 1 components. This is provided by evidence to demonstrate that these welds will be subject to qualified manufacturing inspections, supported by an elastic-plastic fracture assessment to demonstrate tolerance as described for VHI components in the following sub-section. The principle difference between HI and VHI components is in regard to the diversity of evidence to support the arguments, which for HI components will not be as great as that provided for VHI components. In addition to the structural reliability justified by compliance with the ASME B&PV Code, these additional measures are claimed to support a frequency of failure commensurate with HI classification.

VHI Components

As described previously, the structure of VHI component safety reports will be based on five claims to identify diverse evidence to demonstrate the avoidance of significant defects; that any identified defects are tolerable, and that defect growth will not compromise safety through their life-cycle. The numerous arguments and diverse evidence to substantiate these claims will be provided in a separate appendix for each component that includes regions classed as VHI. A general summary of the expected content is provided below.

The first two claims are intended to establish high quality through good design and manufacture, supplemented by functional testing 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. The basis for these claims is established, in part, by compliance with relevant sections of the ASME B&PV Code and informed by relevant operating experience. As such, claims 1 and 2 share some common features with the approach for Class 1 components.

Supplemental measures will be applied to support these claims for VHI components. These will include alternative fracture toughness testing to establish a valid fracture toughness reference temperature for the defect tolerance assessment and for some VHI components, additional stringent control of chemical composition will be specified to minimise the effect of irradiation embrittlement or thermal ageing. High reliability manufacturing inspections will be applied to the VHI components, for which the inspection system, including procedure, equipment, and personnel, will be qualified according to the European Network for Inspection & Qualification (ENIQ)-based methodology for qualification of non-destructive testing. The approach for inspection qualification is described in [Ref-9]. These qualified inspections provide high confidence in establishing the absence of significant defects at the end of all manufacturing stages, before the start of operation, and also generate a set of benchmark data against which future ISI results can be compared. Claim 3 will detail the hydrostatic testing, conducted in accordance with requirements of the ASME BP&V Code.

The fourth claim is that the VHI 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. Defect tolerance is to be demonstrated by fracture assessments to establish tolerance to all defects smaller than a qualified examination defect size by a size margin of two. For VHI components, the elastic plastic fracture mechanics methodology of the R6 [Ref-10] procedure will be used for defect tolerance assessment, as described in [Ref-11]. Evidence will be provided to identify how pressure-temperature limits are prescribed and controlled to prevent rupture, particularly at low temperatures during operation.

The fifth claim is that systems are provided to effectively forewarn of failure. Early indication of degradation is provided to prompt corrective action before gross failure will occur. This is achieved by specification of ISI 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, it will be specified for the VHI locations in accordance with the robust ENIQ-based qualification methodology, as described in [Ref-12]. 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 will provide samples for testing of mechanical properties, fracture toughness and corrosion resistance properties to account for the effects of irradiation embrittlement and thermal ageing.

Materials and Degradation

The component safety reports will include a summary of the evidence provided to justify the structural integrity of those UK ABWR components included within the scope of this sub-chapter for a 60-year period of operation. In order to do this, it is necessary to take account of the potential for degradation. It is planned that a report will be produced in a future step of the GDA to establish that appropriate materials have been selected and that these are compatible with the environment in which they will operate. This will provide a reference to support the detailed description of measures to be used to prevent and monitor degradation, which will be included in the component safety reports in a later step of the GDA.

Operating experience (OPEX) provides a valuable source by which understanding of susceptibility to degradation is developed and maintained. The UK ABWR benefits from this by including design enhancements to decrease the potential for degradation, as compared with other Light Water Reactor (LWR) designs. [Ref-1] describes the origin of the ABWR design.

Particular components of earlier BWR plants have experienced degradation by various mechanisms, notably by stress corrosion cracking (SCC). [Ref-13] describes the UK ABWR strategy for avoidance of 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 UK ABWR will operate with a water chemistry regime intended to prevent degradation. The measures described in [Ref-13] 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, in-service inspection will form an important element for control of degradation through periodic monitoring, particularly at locations where OPEX indicates vulnerability.

8.7 Load Condition

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. In addition, the plant events are classified into the plant operating conditions: Operational Condition I, II, III, IV, Design Extension and Test Condition.

8.7.1 Plant Operating Condition

Postulated events that the plant will or might credibly experience during design life are considered, to establish the design condition for mechanical systems and components. These events are classified into six plant conditions determined in chapter 5.2.2.

8.7.2 Service Load

The plant events are divided into two groups; plant operating events during which thermal-hydraulic transients occur, and dynamic loading events due to accidents, earthquakes and certain operating conditions. The service loads, which are normal loads and dynamic loads induced by these events, are outlined as follows:

(1) Normal Loads
- Dead loads
- Live loads:

The loads induced by movable equipment loads.
- Pressure loads:

Lateral and vertical pressure of liquids.
- Thermal loads:

Thermal effects and loads during plant operating events.
- Reaction loads:

Piping and equipment reactions during plant operating events.

(2) Dynamic Loads [Ref-14]
- Turbine Stop Valve Closure (TSVC), Safety Relief Valve (SRV) Loads:

Hydrodynamic loads induced by pressure waves, due to valve actuation such as TSVC and SRV actuation.
- LOCA Loads:

Hydrodynamic loads and Reactor Building Vibration (RBV) loads induced by LOCAs.
- Seismic Loads:

RBV loads due to earthquakes.

8.7.3 Load Combinations

Load combinations are associated with normal operation, postulated accidents, specified earthquake and other RBV events. Therefore, the load combinations are appropriately considered to evaluate structural integrity. The principles of the load combination for mechanical systems and components are shown as follows:

(1) The loads induced in an event are combined.

(2) When independent events occur simultaneously in certain probabilities, the loads from each event are combined.

(3) The combination of a hazard and the hazard-dependent event is considered in the frequency of the initiating hazard.

8.8 Conclusion

This chapter describes how the structural integrity of metal SSCs that are significant to nuclear safety is to be assured for the UK ABWR. The process to justify structural integrity commences with the establishment of the safety functions required of a particular component, following which, a system of structural integrity classification is applied, to determine suitably rigorous measures that will be required to provide assurance of integrity. This system is based on postulated structural failure of the component with associated loss of its safety function(s), taking account both of the direct and indirect consequences of failure, the degree of redundancy and diversity and availability of protection. The method to establish SFRs and for structural integrity classification are described in Sub-chapter 8.4 and 8.5 respectively.

Component safety reports will be provided in a future revision of this chapter. The structure and content of these will vary according to their structural integrity classification, as described in Sub-chapter 8.6. The nature and extent of evidence necessary to justify an appropriately high level of structural reliability has been summarised for all classes of component which 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. Finally, the approach to specify load conditions and their combination for input to assessments that will support the structural integrity safety case has been described in Sub-chapter 8.7.

This chapter is presented at Step 2 of GDA and its content reflects the current state of development in terms of design substantiation in a UK context; future revisions of this chapter and its appendices will identify detailed information which is currently being developed to substantiate the structural integrity safety case.

8.9 References

[Ref-1] “Genesis of ABWR design” (GA91-9901-0011-00001 Rev.A)

[Ref-2] “Summary of the Design of Reactor Pressure Vessel for UK ABWR” (GA91-9201-0003-00035 Rev.0)

[Ref-3] “Summary of the Design of Main Steam Piping for UK ABWR” (GA91-9201-0003-00036 Rev.0)

[Ref-4] “Summary of the Design of Feedwater Piping for UK ABWR” (GA91-9201-0003-00037 Rev.0)

[Ref-5] “Summary of the Design of Main Steam Isolation Valve for UK ABWR” (GA91-9201-0003-00038 Rev.0)

[Ref-6] “Preliminary Safety Report on Structural Integrity” (GA91-9901-0005-00001 Rev.C)

[Ref-7] “Structural Integrity Classification Procedure” (GA91-9201-0003-00054 Rev.0)

[Ref-8] “Categorisation and Classification of Structures, Systems and Components” (GA91-9901-0007-00001 Rev.B)

[Ref-9] “Inspection Qualification Strategy” (GA 91-9201-0003-00057 Rev.0)

[Ref-10]R6 Revision 4, Assessment of the Integrity of Structures Containing Defects, EDF Energy.

[Ref-11] “Defect Tolerance Assessment Plan” (GA91-9201-0003-00056 Rev.0)

[Ref-12] “Outline of the PSI and ISI Plan for UK ABWR” (GA 91-9201-0003-00039 Rev.0)

[Ref-13] “UK ABWR - Approach for the Avoidance of SCC” (GA11-1001-0003-00001 Rev.0)

[Ref-14] “Preliminary Safety Report on Civil Engineering and External Hazards” (GA91-9901-0004-00001 Rev.B)