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UK ABWR Generic Design Assessment

Preliminary Safety Report on Structural Integrity

Hitachi-GE Nuclear Energy, Ltd.

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Acronyms

Abbreviations and Acronyms	Description
ABWR	Advanced Boiling Water Reactor
AGR	Advanced Gas-cooled Reactor
ALARP	As Low As Reasonably Practicable
ANSI	American National Standards Institution
API	American Petroleum Institute
ASME	American Society of Mechanical Engineers
BS	British Standard
BOP	Balance of Plant
BoSC	Basis of Safety Case
BPVC	Boiler & Pressure Vessel Code
CR	Control Rod Drive
DAC	Design Acceptance Confirmation
DBA	Design Basis Accident
DBE	Design Basis Earthquake
ENIQ	European Network for Inspection & Qualification
FMECA	Failure Modes and Effects Criticality Analysis
GDA	Generic Design Assessment
HI	High Integrity
ISO	International Organization for Standardization
ISI	In-Service Inspection
IoF	Incredibility of Failure
JSME	Japan Society of Mechanical Engineers
LBB	Leak-Before-Break
LEFM	Linear Elastic Fracture Mechanics
LWR	Light Water Reactor
MSLs	Main Steam Lines
NDE	Non-Destructive Evaluation
NPP	Nuclear Power Plant
PCSR	Pre-Construction Safety Report
PSI	Pre-Service Inspection
QEDS	Qualified Examination Defect Size
RIP	Reactor Internal Pump
RPV	Reactor Pressure Vessel
SAPs	Safety Assessment Principles
SCC	Stress Corrosion Crack
SFR	Safety Functional Requirement
SI	Structural Integrity
SSC	Systems, Structures and Components
TAGSI	Technical Advisory Group on Structural Integrity
VHI	Very High Integrity

1. Introduction

The demonstration of structural integrity is a key technical area for the United Kingdom (UK) Generic Design Assessment (GDA) process. Hitachi-GE Nuclear Energy, Ltd (Hitachi-GE) recognises that there are important differences between the UK Regulatory expectations and those adopted by other international regulators. This report has been written to highlight some of the important issues and to outline the strategy for demonstrating structural integrity of the UK Advanced Boiling Water Reactor (UK ABWR).

In the UK, it is necessary to provide a safety case to demonstrate how the structural integrity of nuclear power plant systems, structures and components (SSC) can be assured over the design life of the plant to a level of structural reliability and a degree of rigour commensurate with the consequences of postulated gross failure.

To achieve this, in the first instance, a classification methodology is required, based on consideration of the direct and indirect consequences of postulated gross failure. Based on this classification, it is possible to present structured arguments for each major component and tailored to support the structural reliability claimed for that component. There are some structures or components within nuclear power plants where the radiological consequences of gross failure are intolerable and where physical safeguards or barriers cannot provide protection and it needs to be shown that the likelihood of gross failure of a particular structure or component is so low that it can be discounted (i.e. it is incredible). For such components, there is a requirement to provide arguments and evidence to support the claimed high levels of structural reliability (nominal probability of failure of $<10^{-7}$/year) and substantiating such claims requires a high burden of proof and often requires measures to be taken over and above established nuclear design code requirements.

The purpose of this document is to describe Hitachi-GE's proposed strategy for the structural integrity classification of Advanced Boiling Water Reactor (ABWR) structures and components and to provide the strategy to substantiate the structural reliability claims for components. The purpose of this document is also to outline a provisional programme of work planned during the subsequent stages of the GDA process. Hitachi-GE proposes to adopt the term Very High Integrity (VHI) for those components where the highest reliability requirements apply. This is equivalent to the term 'Incredibility of Failure' (IoF) which is used by some UK utilities.

1.1 UK ABWR Programme and GDA Process

Hitachi-GE is working with the support of Horizon Nuclear Power to put the ABWR design through the GDA process in the UK. The intention is to build some units across two sites at Wylfa, Anglesey, Wales and Oldbury, South Gloucestershire, England, with the first units becoming operational in the mid-2020s. The ABWR is an established design which is licensed in Japan, Taiwan and the United States of America (USA). The first plant began commercial operation at Kashiwazaki-Kariwa in Japan in 1996. Since then three additional plants have commenced operation in Japan with others under construction in Japan, and two in Taiwan.

The purpose of the Office for Nuclear Regulation (ONR) GDA process is to provide a structured process for early engagement between the ONR and Hitachi-GE, primarily to allow the ONR to evaluate the safety cases for the UK ABWR well ahead of any construction in the UK, and for any safety issues to be identified and resolved to ONR's satisfaction early in the project. Hitachi-GE's aim is to provide the ONR with sufficient information relating to the ABWR design to allow the ONR to undertake a thorough and detailed assessment of the information provided in the relevant topic areas, including Structural Integrity. Hitachi-GE's overall objective of the GDA is to provide sufficient confidence to the ONR to permit a Design Acceptance Confirmation (DAC) to be issued, enabling site specific licensing to commence. The GDA process is staged with the level of technical information and the depth of assessment increasing with each step.

The key steps are as follows:

- Step 1: Preparation for GDA.
- Step 2: Fundamental safety overview – an overview of the fundamental acceptability of the ABWR design within the UK regulatory framework.
- Step 3: Overall design safety review – ONR review of the safety aspects of the ABWR design.
- Step 4: Detailed design assessment – ONR carry out an in-depth assessment of the safety case.

This document has been prepared in support of the Step 1 submission. The purpose of this report is not to provide detailed technical information of the ABWR design; rather it is intended to outline the overall strategy for establishing the structural integrity classification of components and the proposed approach for providing justification of the claimed levels of structural reliability. The intention is to provide confidence that the Hitachi-GE strategy aligns with ONR expectations and to enable early discussion and resolution of potential concerns.

1.2 Hitachi-GE Approach to Structural Integrity

The ABWR is an established and proven design with licenses issued in Japan, Taiwan and the USA. The plant has been designed and manufactured in accordance with both the Japan Society of Mechanical Engineers (JSME) and American Society of Mechanical Engineers (ASME) design code requirements. The basic requirements in terms of component and piping design assessment, material selection, manufacturing quality, manufacturing inspection and testing are based on demonstrating conformity with the design code requirements, and other nationally prescribed regulatory requirements. The robustness of the ABWR plant is also supported by many years experience in the design, manufacture and operation of BWRs worldwide. The design has continually evolved to improve reliability and safety and this experience is embedded into the ABWR approach to structural integrity. Hitachi-GE recognises that in the UK, the overall approach to regulation is different in so far as the Regulatory approach is ‘permissive’ rather than ‘prescriptive’. In order to demonstrate consistency with the ONR Safety Assessment Principles (SAPs) measures over and above the design code requirements will be required for certain components. Hitachi-GE also recognises the important role that the safety case has in the UK, in terms of explaining the basis of the safety argument, and providing the evidence to underpin the safety case claims. This strategy document includes an explanation of Hitachi-GE’s proposed strategy for the demonstration of structural integrity.

1.3 Objectives of the Structural Integrity Strategy

Hitachi-GE understands the importance of establishing a clear strategy for ensuring that the ABWR is compliant with the UK structural integrity requirements. Evidence from previous GDA assessments in the UK has highlighted that for components where it needs to be shown that the likelihood of gross failure of a particular structure or component is so low that it can be discounted can be an area of technical risk in relation to the successful outcome of the GDA process and achievement of the DAC. This risk arises either because the proposed approaches do not fully align with UK Regulatory expectations, or because the level of work required for the necessary assessments cannot be realised within the GDA project timescales, meaning that the evidence available to the ONR was late or incomplete.

In mitigation, Hitachi-GE therefore intends to establish the strategy for structural integrity early in the GDA process. The objective of this document is to outline the strategy proposed by Hitachi-GE to achieve this goal. This document represents an interim position which will continue to be developed throughout the GDA process and has been prepared as a result of a technical meeting between the ONR and Hitachi-GE in October 2013 where the outline strategy was discussed. Further development of the strategy will continue into the implementation phase which will commence with GDA Step 2.

The ONR SAPs do not prescribe how the structural integrity classification of a component should be established or what the required evidence to substantiate the claimed structural reliability should consist of. This document therefore seeks to outline Hitachi-GE's proposal for satisfying the ONR SAPs as applicable to the UK ABWR design.

Fundamental to the UK ABWR component structural integrity strategy is the clear understanding of the potential radiological consequences of any postulated failure mode. Accordingly, a structural integrity classification methodology is being developed. This classification scheme will provide 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 the structural reliability claim will be based on the demonstration of good design and build in accordance with recognised nuclear design standards is appropriate. For these components, the applicable codes will be established on the Quality Class of the component.

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 need to be substantiated.

1.3.1 High Level Programme

The proposed programme of work for the demonstration of structural integrity is linked to the GDA steps, whereby the level of detail will progressively increase with each of the GDA steps. However, Hitachi-GE is aware that previous GDA projects have suffered from late regulatory assessment in some technical areas which reduced the overall benefit to the requesting party and resulted in many GDA issues arising at the end of Step 4. To mitigate this risk, Hitachi-GE therefore intends to bring some of the GDA Step 3 work forward into GDA Step 2 with a view to the following:

i) Establishing the overall structural integrity strategy during GDA Step 1,
ii) Providing overview of approach to management of Stress Corrosion Cracking (SCC) during GDA Step 1,
iii) Defining the classification scheme and method of substantiation during GDA Step 2,
iv) Defining a process for prioritising weld locations for defect tolerance and NDE qualification assessment during GDA Step 2,

v) Identifying the candidate VHI and HI components during GDA Step 2 and commencing the defect tolerance assessment during GDA Step 2 for the most onerous welds,
vi) Commence programme study for identifying Structural Integrity (SI) classification of Balance of Plant items during GDA Step 2,
vii) Providing overview of approach to inspection qualification during GDA Step 2,
viii) Commence NDE evaluations during GDA Step 3,
ix) Starting the development of the safety case arguments prior to completion of GDA Step 3.

Hitachi-GE considers that this approach will support timely resolution of ONR queries, observations and issues and ultimately help achieve a reduction in the overall GDA timescales.

2. Structural Integrity Classification Process

2.1 Introduction

A key principle of the UK ABWR structural integrity strategy is that the level of rigour applied to the structural integrity justification is related the nuclear safety consequences of failure. The component classification methodology provides a structured approach for both identifying which components need to be considered against the highest reliability requirements and for which components a consequences argument can be made. In applying the methodology, the consequences of failure will be considered in a broad sense, and include the direct effects on plant safety, for example, loss of coolant inventory or reactivity excursions, and the indirect effects from, for example missiles, blast, pipe whip, flooding and water/steam jets. This will also consider any longer term effects on the availability of essential safety systems and instrumentation required to maintain plant safety in fault conditions. For the ABWR, the assessment will need to consider the Balance of Plant as well as the nuclear island.

For such consequence assessments, Hitachi-GE recognises that UK industry practice is to assume a gross failure irrespective of any inherent mitigation arguments based on anticipated material properties. Furthermore, in the absence of either engineering features, restraints or other design features to limit the consequences of failure, Leak-Before-Break (LBB) is not normally used as a basis for substantiating a limited consequence argument.

Thus, in pipework assessment, a guillotine failure will be assumed. The tolerance of the plant to the postulated guillotine break will then be assessed though consideration of the capability of the ABWR protection systems. The indirect consequences arising from pipe whip or jet impingement will also be considered. The approach may be applied in a qualitative sense, for example through inspection of the plant layout or CAD models and consideration of the proximity and safety significance of adjacent systems. If engineering judgments are less straight forward to substantiate, a more detailed evaluation may be required, involving a pipe whip assessment using an impact assessment procedure. Similarly, for vessels, gross failure will be considered along with the generation of missiles, which may subsequently threaten the next barrier, e.g. the containment. By following this logic, it falls to the safety case to provide assurance that, for example, vessel material will behave in a ductile manner for the design life of the plant, or that environmental effects could not lead to a guillotine failure of a pipe. These arguments will provide defence in depth to the safety case.

2.2 Proposed Component Classification Scheme

The safety functional requirements of structures, systems and components (SSCs) will be identified from the UK ABWR safety schedule. The appropriate safety categorisation of functions and classification of components will be determined in accordance with ONR SAPs ECS.1, ECS.2 and associated paragraphs. Hitachi-GE recognises that it is important that the structural integrity classification methodology is consistent with the overall UK ABWR safety classification scheme which is currently being developed. Based on the preliminary work, it is expected that this will define three broad classes for SSCs, namely: Class 1, Class 2 and Class 3, which are discussed in the Step 1 document [Ref-5] titled ‘Categorisation and Classification of Structures, Systems and Components’ in detail.

In practice, this Classification methodology is too coarse to determine component structural integrity requirements, since Class 1 covers a very broad range of initiating event frequencies and consequences and hence a commensurately broad range of structural reliability targets. Hitachi-GE proposes to use an alternative classification methodology for determining structural integrity requirements for passive structures which sub-divides Class 1 SSCs according to the consequences of gross failure. Although no prescriptive methodology is available, one such example, presented in

[Ref- 1], is representative of classification schemes developed within a UK regulatory context and has been selected as the basis of the this approach.

2.2.1 Very High Integrity (VHI)

Very High Integrity (VHI) is assigned to failure modes for which there is no protection and failure is intolerable and where it is not reasonably practicable to provide protection. For such components the frequency of gross failure should be judged to be below 10^{-7}/year. In practice this applies to fault sequences resulting in both severe core damage and large uncontrolled release. [Ref- 2], prepared by the UK TAGSI (Technical Advisory Group on Structural Integrity), provides an authoritative view on the requirements for VHI components and how the safety case for such components should be presented. This has been used in the development of this document. For Very High Integrity components, Hitachi-GE proposes that the presentation of the safety case arguments will broadly follow the format described in [Ref- 2], modified to align with the Claim-Argument-Evidence structure. A defect tolerance study will be undertaken to satisfy SAPs EMC.1 and EMC.5, and, for VHI components it is planned to substantiate the claim through the application of qualified inspections to provide a high level of confidence that the inspection technique is able to detect structurally significant defects with a high level of reliability. The detailed approach for the justification of VHI components will be developed in the next stages of the GDA.

2.2.2 High Integrity (HI)

The failure of High Integrity components can lead to severe core damage but in this event effective containment exists to limit the off-site consequences. Hitachi-GE recognises that the structural integrity requirements for components in the High Integrity category remain onerous as their significance to safety is still very important. The TAGSI structure will be adopted for HI components, whilst comprehensive evidence will be presented, for some arguments, the depth of the evidence will not be as rigorous as for VHI components.

High Integrity will be assigned where the component can be described as single failure intolerant, i.e. where this is a single line of protection, but no redundancy. In this case, it is therefore possible to make a consequence argument, but this will not provide the same degree of confidence from a fully qualified line of protection with redundancy. Thus, partial protection will be available and this will be reflected in the degree of substantiation that will be provided. In this case, a full justification of the claims for the consequence arguments will be required.

The High Integrity classification covers a broad band of reliability targets (10^{-7}/year < and <10^{-5}/year). This follows from the degree of protection available and a judgement will be made as to whether the Tolerable Failure Frequency target should be biased towards the upper or lower end of the band. If severe core damage is inevitable or if there is the potential for significant secondary damage then the target will be a failure rate < 10^{-6}/year. Conversely, if there is confidence in the protection claim for the majority of the load cases, Hitachi-GE will aim to achieve something closer to code compliance (<10^{-5}/year). For both situations a defect tolerance study will be undertaken, and the subsequent inspection requirements will depend on the outcome of such studies. In situations where small tolerable defect sizes need to be defended then targeted qualified inspections may be required.

As with VHI cases, compliance with the design and manufacturing requirements of ASME provides the basic building block through ‘Achievement of Integrity’ for the lower integrity classifications. Additional controls on materials and fabrication may be necessary. Furthermore, a ‘Demonstration of Integrity’ will be provided for a component classified as High Integrity. The detailed approach for the justification of HI components will be developed in the next stages of the GDA.

Hitachi-GE understands that this type of approach is used in the UK Advanced Gas-cooled Reactor (AGR) fleet. Noting that the UK classification methodology has not previously been applied to a

BWR design, Hitachi-GE proposes to retain the High Integrity classification, although this position will be kept under review once the proposed methodology has been applied.

2.2.3 Standard Class 1

It is required that only limited core damage results in the event of failure of components classified as Standard Class 1. And it is also required that they have at least one line of protection, with redundancy. The integrity claim for such components will (in the absence of a known degradation mechanism) be based on compliance with recognised nuclear design code requirements. Thus, the safety case for components in this category will be made on the basis of the quality of the design and build. A multi-legged safety case argument will not be presented for Class 1 components.

2.2.4 Standard Class 2 (and below)

Failure of components classified as Standard Class 2 should not result in any core damage. And it is also required that they have at least two lines of protection, with diversity. The safety case will rely on the use of appropriate design codes and standards. Standard NDE in accordance with code requirements will be used to underpin integrity claims. Specific measure may be applied where known degradation mechanisms exist.

2.2.5 ALARP

The ALARP (As Low As Reasonably Practicable) principle will be applied by Hitachi-GE to minimise plant risk to a level that is as low as reasonably practicable through measures such as the careful selection of materials and manufacturing processes, defining inspection requirements and undertaking materials surveillance. For VHI and HI components, Hitachi-GE will consider whether reasonable measures could be taken to mitigate the consequences of failure.

2.2.6 Seismic Classification

The seismic design of the NPP provides mitigation in relation to the radiological hazard by maintaining the integrity of SSCs during and after an earthquake. Therefore, the seismic category that corresponds to radiological dose (the off-site consequences) is defined, and the integrity of SSCs is evaluated by use of the earthquakes corresponding to each seismic category. The details are as follows:

Seismic Category 1: SSCs whose failure would lead to off-site dose > 10mSV and which must therefore confirm integrity during and after DBE.

Seismic Category 2: SSCs whose failure would lead to off-site dose > 0.01mSV and which is therefore designed to confirm integrity during and after 10^{-3}/year earthquake.

Seismic Category 3: SSCs whose failure would lead to off-site dose < 0.01mSV and required for operation. They are designed by nominal industrial seismic design standards.

Moreover, SSCs that are not classified as the highest seismic category (Seismic Category 1) but which satisfy the following conditions are classified as Seismic Category 1A.

- SSCs whose failure could lead to the failure of an adjacent Seismic Category 1 SSC;
- SSCs which are needed to perform a Beyond Design Basis Accident (BDBA) function after an earthquake or other hazard events.

The details for the seismic categorization and seismic design motions are described in the Step 1 document [Ref-6] titled ' Preliminary Safety Report on Civil Engineering and External Hazards'.

2.3 Methodology for identifying VHI and HI Components

The proposed classification scheme has been selected to ensure consistency with the overall ABWR Categorisation and Classification process. Components assessed as VHI and HI are essentially a sub set of Class 1 SSCs.

Because it is not practical to review every component on the plant in the identification of candidate VHI and HI components, the starting point of the process is a review of all the Class 1 SSCs. Based on a level of technical judgement, a shortlist of components for further assessment will be identified. As a minimum this review will include all components on the reactor coolant pressure boundary within containment, including the RPV, the MSLs, main steam isolation valves, feed water lines, the reactor recirculation system (the Reactor Internal Pump), the residual heat removal system and heat exchangers, the high pressure core flooder system and others. The Reactor internals will also be considered. A selection of BOP components outside containment will also be considered.

The approach for the assessment of safety classification will be based on the Failure Modes and Effects Criticality Analysis (FMECA) technique. This will provide a logical framework to conduct the assessment. It is proposed that the first assessment will be based on an expert panel approach, involving specialists in structural integrity, fault studies, plant performance, internal hazards, as well as UK expectations. The FMECA will be used to summarise the expert review and to identify evidence to underpin the claims. Once the candidate VHI and HI components have been identified, it is necessary to identify subregions on the component for evaluation in the FMECA. For vessels, these subregions will include major welds, regions of structural discontinuity and nozzles. In piping, these will include an assessment of the limiting welds. This may require an evaluation of multiple locations, particularly where secondary consequences are of concern. If the number of locations becomes excessive, they will be rationalised on the basis of, for example, material properties, operating stresses and transient loadings.

Depending upon the number of components being assessed, subregions may be considered at a later stage, once the VHI items have been identified at a component level. The benefit of this approach is that whilst the RPV may be assessed as VHI, there may be some locations on the vessel where the full requirements of VHI need not apply, these can only be identified using a detailed review of subregions.

Having identified the subregions for assessment, the next step is to postulate a gross failure of that region, for a pipe, this may consider a guillotine break, for a vessel weld; this may consider a non-ductile failure of the shell. An important aspect of this is that, in general, gross failure will be postulated irrespective of the anticipated material behaviour. This means that no direct benefit will be claimed from anticipated Leak Before Break (LBB) behaviour, or the assumed ductility of the pressure vessel material. For nozzles attached to the RPV pressure boundary, major structural welds will consider the consequences of failure of the RPV shell, however, for minor penetrations (such as instrumentation lines) which are attached to the cladding using a non-structural weld, the consequences will be assessed on the basis of a guillotine break of the penetration.

For the postulated gross failure, the direct radiological consequences will be evaluated. This will consider firstly the mitigation degraded consequences of failure, for example the potential effects of loss of core cooling, or reactivity excursions, and the effectiveness of containment. Next the effectiveness of protection systems to mitigate the failure will be evaluated. This will involve appropriate interactions between specialists in ABWR fault studies and structural integrity.
Following on from the consideration of the direct consequences of failure, the indirect consequences of failure will be assessed. This will consider the effects of missile generation, pipe whip, blast, jet impingement and containment over-pressurisation. For each component, consideration will be given to the location of the component on the plant relative to other Safety Related equipment, such as adjacent component, piping systems and control and instrumentation systems. Consideration will be given to the effectiveness of internal barriers to provide protection to adjacent compartments.

The initial review will be qualitative, based on a review of the plant general arrangement layouts or 3D CAD models, and some engineering judgment will be applied. For some locations, where the judgements are less robust, a more in depth study will be required. For piping systems, this will need to consider the effectiveness of pipe supports and restraints, the locations where plastic hinges could form in a pipe, the proximity of the whipping pipe to adjacent systems and the target response to the pipe impact. The radiological consequences of the secondary consequences then need to be evaluated. Whilst the initial failure will be assumed to be a gross failure, any secondary failures will be assessed on a best estimate basis. For example, pipe whip on failure of an adjacent system may not necessarily result in a guillotine break in both systems. Finally, based on the assessment of the direct and indirect consequences of failure, the safety classes will be assigned.

3. Avoidance of Failure

3.1 Objectives

The demonstration of defect tolerance provides an important contribution to any claim of high structural reliability. In combination with the use of high reliability NDE techniques, it is possible to provide a significant contribution to a VHI or HI claim where it can be shown that the NDE techniques applied during manufacture have a high probability of detecting defects at a size below which they need to be considered as being structurally significant. The objective of the defect tolerance assessment is therefore to establish, to a high level of confidence, the size of planar defect that needs to be considered as being structurally significant at the start of life. This information will then be used to inform the design, qualification and implementation of the NDE systems applied during manufacture.

Conventionally, the defect tolerance assessments undertaken on the ABWR are based on the Linear Elastic Fracture Mechanics (LEFM) approach described in Section III, App G of the ASME Boiler and Pressure Vessel Code. Hitachi-GE understands that this approach is not considered sufficiently robust in the context of UK regulatory expectations. Hitachi-GE therefore proposes to apply the R6 [Ref- 3] elastic plastic fracture mechanics assessment technique as supplementary. The R6 methodology has been selected because it is well established in the UK, it has been validated for use within the UK nuclear industry and it is recognised by the UK ONR.

3.2 Selection and Prioritisation of GDA Assessment Locations

Completing the final R6 assessment of all the VHI and HI locations during the GDA timescales, may prove to be an onerous task. Given that the overall objective of the GDA process is about progressively building confidence in the design, such that there is confidence that the design is capable of being licensed in the UK, Hitachi-GE proposed to follow a staged approach to the defect tolerance assessment and NDE evaluation. This approach is based on the selection and prioritisation of the limiting locations for assessment and focussing efforts on these initially to provide overall confidence in the defect tolerance of the plant during normal, upset and faulted load conditions.

To identify the bounding ABWR locations for assessment, Hitachi-GE proposes to undertake a ranking based on a qualitative assessment of the defect tolerance of each location and the inspectability of each location. The prioritised list of locations based on the NDE ranking and defect tolerance ranking will be reviewed to ensure that it includes a suitable representation of each weld type and component.

The limiting locations will initially be selected for assessment (e.g. the RPV is the most obvious example) during Step 2, and arguments made enabling the results of the bounding locations to be read across to other locations, such that all VHI and HI locations can be evaluated within GDA timescales and the necessary assurance provided.

3.3 Approach to the Application of R6 methodology

The overall approach to the application of R6 methodology will be to determine the critical defect size at end of life, i.e. the defect size that just lies outside the R6 failure assessment diagram. Then, by undertaking a reverse fatigue crack growth assessment, it is possible to establish the maximum allowable defect size at start of life.

For the selected bounding locations, defects will be assessed in a variety of locations with appropriate material properties selected for all materials, both parent and weld. Finite element analyses will be used to determine the primary and secondary stresses for the range of transient load cases and used in both the fatigue crack growth assessments and the R6 assessment.

3.4 Materials

Materials for metal components and structures will be selected for the purpose of enabling the UK ABWR to be designed, manufactured, operated, inspected and maintained in accordance with regulatory expectations and thought of ALARP. In selecting materials, it is vital that for the materials selected will be able to maintain the reliability and safety of the SSCs with adequate margin (addressed in the safety case) over the design life of the plant. The selection of materials should thus consider the material properties of each material based on the design specifications of each SSC with due care and attention to the environment in which the material is situated.

Carbon steels will in principle be used for metal components, but will not be selected where usage and environmental requirements deem it to be unsatisfactory, for example in temperature regions where degradation due to creep could be an issue. For each material selected, the corrosion allowances shall be decided upon by consideration of purpose, environment, design life of each component, etc. For plant components in direct contact with liquids, selection of the materials chosen will consider properties of the fluid, dissolved oxygen, operating temperature, flow velocity etc. Alloy steels, such as low alloy steel and stainless steel, to which alloy elements are added, are selected instead of carbon steel in consideration of the ambient conditions of the material. The corrosion of carbon steel pipework will be minimised by use of oxygen injection or inhibitors, of which further detail will be provided in future GDA submissions.

Stainless steel (austenitic stainless steel) shall be used where appropriate, but not in very high temperatures due to consideration of creep degradation phenomenon. Moreover, stainless steel of low carbon specification having the resistance to heat sensitization shall be taken into consideration for use in equipment and pipes with welded construction in the system contacting water and steam at high temperature environment.

It is the current intention that the reactor internal structures (Steam Dryer, Steam Separator, Core Shroud, CR tube, Top Guide, Core support, CRD housing, etc.) shall largely use low carbon stainless steel which was added stricter specification (nuclear grade) with the superior Stress Corrosion Cracking (SCC) resistance, with the exception of the shroud support structure that requires higher strength, which will use modified nickel-based alloy with the superior SCC resistance. The use of stainless steel for the reactor internals will also reduce the occurrence of corrosion products and level of radiation from the corrosion products. To suppress the radiation dose from the equipment and piping of the primary reactor system, low-cobalt steels will be used to minimise the major dose contribution from ^{60}Co.

3.5 Materials Degradation

Material degradation has become a significant issue for operators of LWRs in the last 30 years. The list below illustrates some of the material degradation issues, which have arisen from BWRs operating experiences:

- Transgranular Stress Corrosion Cracking (SCC) of stainless steel due to chlorides in BWRs.
- Intergranular SCC of austenitic stainless steel and nickel-based alloy in BWRs.
- Irradiation assisted SCC of stainless steel in BWR core internals.
- Flow accelerated corrosion of carbon steel in piping.

- Transgranular SCC of carbon steels in BWR feedwater piping.
- Irradiation Embrittlement of low alloy steel in BWRs (Shown in the 3.5.2 chapter)

These issues will have to be considered for the UK ABWR, and detailed explanation of the management strategies and design changes to mitigate the risks will be discussed in successive steps of the GDA process. These strategies have been shown to be generally effective and the design of the ABWR incorporates many of the lessons learned and mitigative measures that have been developed by Hitachi-GE and other bodies over the preceding decades. The issues of SCC and Irradiation Embrittlement are briefly discussed in the sub-sections below.

3.5.1 Control of Stress Corrosion Cracking

During the 1970's, Stress Corrosion Cracking (SCC) of stainless steel and nickel based alloy in BWR plants became a worldwide concern. In common with all light water reactor designs, the susceptibility to SCC is generally understood to be governed by three factors; material susceptibility, stress level and susceptible environment. Hitachi-GE has developed countermeasures in each of these areas and a detailed explanation of these countermeasures for specific regions of the plant will be provided in subsequent steps of the GDA process.

With regard to the selection of materials, the ABWR plant makes extensive use of niobium stabilised grades of nickel based alloy (e.g. for the core shroud supports and the control rod drive mechanism stub tubes) and low carbon stainless steel. Although the water chemistry for the UK ABWR has not yet been specified, some options are being evaluated such as the application of Hydrogen Water Chemistry in conjection with platinum injection and zinc injection.

The manufacture of ABWR components also utilises many advanced manufacturing techniques selected to reduce susceptibility to SCC, such as the modification of residual stresses and surface cold work through optimisation of manufacturing processes, use of appropriate welding techniques, polishing, peening and induction heating stress improvement. Testing has shown that reduction of surface hardness due to cold work can be an effective at preventing SCC crack initiation. More evidence of an effective strategy to prevent and mitigate SCC in the UK ABWR will be provided in later GDA steps. The summary of basic policies to be considered for SCC in austenitic stainless steels and nickel base alloys based on the past experiences and current studies is described in the supporting document. [Ref-12]

3.5.2 Irradiation Embrittlement

For low alloy steels such as the RPV body (particularly the beltline region), reduction of the effects of irradiation embrittlement is an important design consideration as the risk of non-ductile failure late in life will be increased. It is important to note that as compared to a PWR plant, BWR plant is less severe in the influence at the beltline and less susceptible to pressurised thermal shock transients. Design measures for the avoidance of brittle fracture include minimising stress through careful design, avoidance of stress raising features and reduction of residual stress using post weld heat treatment. The severity of thermal transients on the RPV wall are minimised through the careful design of nozzle, including the use of thermal sleeves.

The ABWR material specification for the RPV beltline has been optimised to reduce the effect of irradiation embrittlement by the supplementary controls on the material specification, for example through the reduction in impurity elements such as Cu, S and P. Start of life properties will be confirmed through a combination of drop weight tests and Charpy impact tests. The requirements for supplementary testing for the UK ABWR will be considered in later GDA steps. In core surveillance specimens will also be included in the design to allow the effects of irradiation embrittlement to be monitored at intervals throughout plant life. Data arising from this surveillance is intended to indicate whether in-service materials toughness remains consistent with values used in determining the end-of-life pressure and temperature limits. Further details will be provided in later GDA steps.

4. Non Destructive Examination

This section summarises the Hitachi-GE's approach to NDE for the UK ABWR and, in particular, in relation to the VHI and HI components. It also describes the approach that will be adopted during GDA to provide confidence in the effectiveness of the approach.

4.1 General approach to NDE

The manufacturing quality of components is assured by performing the NDE required by the design code and augmented by the fabricator's in-house inspections where necessary. This comprises a range of volumetric and surface NDE methods performed at various stages of manufacture. For those components that are classified as VHI and HI, High Reliability (qualified) NDE will be performed at the end of the manufacturing process. This High Reliability NDE will support the avoidance of fracture claims made for these components. Whilst the High Reliability NDE will be performed on VHI components at the end-of-manufacture, qualified NDE may be performed at earlier stages of manufacture if improved access is judged to be beneficial. The following sections summarise the main features of the approach that will be adopted for the High Reliability NDE.

4.2 Scope of NDE for High Reliability Welds

For each of the VHI and HI welds, Hitachi-GE will describe the inspections that are performed at each stage of manufacture and culminating in qualified NDE performed at the end-of-manufacture. Hitachi-GE will show that, on the basis of the NDE performed at the earlier stages of manufacture, there is a very low likelihood of any significant defects remaining at the time of performing the High Reliability NDE.

4.3 Production of Inspection Specifications

Objective based end-of-manufacture NDE will be performed for the VHI and HI welds, meaning that pre-defined inspection requirements will be produced against which the NDE will be designed and qualified. The inspection specifications will include:

- A description of the components to be inspected including all of the relevant features that relate to the performance of the NDE.
- A description of the manufacturing defects that are to be considered for the NDE.
- The performance requirements for the NDE, defining the defect size (based on the QEDS) for which the inspection must be effective and the other essential defect parameters such as location and orientation. QEDS stands for Qualified Examination Defect Size and it is an allowable defect size based on fracture mechanics analysis at start of life with at least margin of two against an inspectable defect size.

4.4 The Basis for the Design of High Reliability NDE

Wherever reasonably practicable, Hitachi-GE will apply ultrasonic methods for the High Reliability end-of-manufacture NDE of VHI and HI welds. The ultrasonic NDE will be designed specifically to detect and assess planar defects for the full range of defect parameters defined in the inspection specification.

4.5 Design for Inspectability

The VHI and HI components of the UK ABWR will be designed to facilitate the end-of-manufacture NDE and in particular, the application of ultrasonic NDE methods.

4.6 NDE Qualification for the UK ABWR

Hitachi-GE will provide additional confidence for the end-of-manufacture NDE through a formal process of qualification. Here the NDE procedures, equipment and personnel will be qualified according to the European Network for Inspection & Qualification (ENIQ)-based Methodology. The principle features of the qualification process will include:

- The production of detailed component specific inspection procedures.
- The production of technical justifications that define the basis of the NDE design and present evidence for the capability and reliability of the NDE to meet the specified inspection requirements.
- The assessment of the NDE procedures and equipment by an independent qualification body. Here the qualification body will assess the technical justification and conduct/manage practical demonstrations of the NDE capability.
- The assessment of the NDE personnel including the conduct of practical blind trials.

4.7 Pre-Service and In-Service NDE for VHI Welds

The UK ABWR will be subject to Pre-Service Inspection (PSI) prior to reactor start-up and In-Service Inspection (ISI) at intervals during operation. The PSI will be performed to use the equivalent basic techniques and equipment that are likely to be applied for the ISI. The ability of the PSI and ISI to meet pre-defined inspection objectives will be assured through a formal process of inspection qualification by applying the ENIQ-based Methodology. The scope of this qualification will include inspection procedures, equipment and personnel.

The scope of the PSI/ISI will be described in the GDA along with a discussion of how the component design facilitates NDE. It is described how PSI and ISI are used to confirm the safety of the Very High Integrity (VHI) and High Integrity (HI) component of UK ABWR at the beginning of the operational life and through the lifetime of the plant in the supporting document. [Ref-11]

4.8 GDA Activities: UK ABWR NDE

During the GDA, Hitachi-GE will make submissions to ONR that will provide more details of the approach to the NDE of the VHI and HI welds that arise out of the classification. The NDE that is performed at each stage of manufacture for the VHI or HI welds that have been identified from the classification will be described. This will show how the NDE establishes the manufacturing quality and provides assurance that there are no significant planar defects at the end of manufacture. During GDA, Hitachi-GE will describe:

- The roles and responsibilities for each of the qualification activities.
- The approach to be used to derive inspection specifications.
- How the inspection procedure is qualified including the use of technical justifications and practical trials for the qualification of the inspection procedure.
- How NDE personnel are qualified (including the use of blind practical trials).
- The arrangements for the qualification body.
- How the qualification of procedures and personnel are qualified.

During GDA, Hitachi-GE will provide evidence, on a sampling basis that the High Reliability NDE performed at end-of-manufacture can meet the required capability using readily available NDE techniques and that these inspections can be qualified. Hitachi-GE will generate the sample to be considered using a ranking process to derive those welds that present the greatest challenges to the NDE. This ranking will combine the output from:

- The preliminary defect tolerance assessment (thereby defining those welds that will have the smallest values of QEDS.)

- The complexity and difficulty of the inspection such as access and materials.

The principle here will be to present overviews of the NDE techniques that might be applied and to provide physical reasoning (qualitative arguments and present available evidence as to why the proposed techniques will be effective.)

Hitachi-GE will also use the GDA process to provide examples of the qualification approach by:

- Using the ENIQ-based format for technical justifications to present the overview of the inspection techniques and physical reasoning.
- Including initial proposals as to how the qualification would be undertaken.

4.9 Assuring the Reliability of NDE for VHI or HI Forgings

Due to the safety significance of the VHI or HI component forgings, Hitachi-GE will assure the reliability of the in-manufacture NDE of these components by producing capability statements and supplement the standard techniques where necessary. The objective of these inspections will be to:

- Ensure that these items enter service free of defects of structural concern.
- Demonstrate that there are no defects in the VHI or HI forgings that could significantly impede the High Reliability end-of-manufacture NDE or the PSI/ISI (pre-service inspection / in-service inspection).

5. Applicable Codes and Standards

In this section, the proposed codes and standards to be applied for structural integrity components in the UK ABWR are outlined, but it needs to be stressed that the allocations proposed here are preliminary and will only be finalised once the classification process of SSCs has been completed. The detailed explanation of Applicable Codes and Standards is provided in the Step document [Ref-4] titled ‘Approach to the adoption of Codes and Standards for the UK ABWR.’

Quality classification of SSCs is an important part of Japanese practice, and further consideration will be needed as to its applicability in the UK and has been included in Appendix A for information purposes.

5.1 Class 1 Components

For pressure retaining components whose integrity is important to safety, it is the intention of Hitachi-GE that they are designed in accordance with the ASME Boiler & Pressure Vessel Code Section III, Division 1, specifically subsections NB, NC, ND and NG. Table 1 identifies the major codes and standards provisionally allocated to be applied for class 1 components in the UK ABWR.

5.2 Class 2 and 3 Components

Class 2 components for the NSSS will be designed in accordance with ASME Boiler & Pressure Vessel Code Section III, Division 1, specifically subsection ND. For BOP items, the current proposal is to use ASME Boiler & Pressure Vessel Code Section VIII, although this may be amended following the completion of the classification process for BOP SSCs.

Class 3 components will be designed in accordance with ASME Boiler & Pressure Vessel Code Section VIII with ISO, European and BS standards also used on some components, and explicitly stated where applied. Piping and valves will be designed in accordance with ANSI/ASME B31.1 and B16.34 to maintain consistency with ASME Sec. III piping.

Table 2 identifies the major codes and standards provisionally allocated to be applied for class 2 and 3 components in the UK ABWR.

Table 1 Structural Integrity Codes and standards for Class 1 Components

SSC Type	Applicable Codes and Standards
Reactor Pressure Vessel	ASME BPVC Section III, Division 1 ASME BPVC Section II, ASME BPVC Section V, ASME BPVC Section IX, ASME BPVC Section XI
Heat Exchangers	
Storage Tanks	
Valves	
Piping	
Pumps	
Supports	
Reactor Internals	
Primary Containment Vessel	ASME BPVC Section III, Division 1 ASME BPVC Section III, Division 2
Pool	ASME BPVC Section II, ASME BPVC Section V, ASME BPVC Section IX

Table 2 Structural Integrity Codes and standards for Class 2 and 3 Components

SSC Type	Applicable Codes and Standards
Pressure Vessels	ASME BPVC Section VIII, Division 1and 2
Heat Exchangers	ASME BPVC Section VIII, Division 1and 2
Piping	ASME BPVC Section VIII, Division 1and 2 ANSI/ASME B31.1 BS EN 13480
Pumps	BS EN ISO 13709 Hydraulic Institute Standards BS Pump Manufacturers Association API 610
Valves	ANSI/ASME B31.1 ANSI/ASME B16.34
Storage Tanks	BS EN 14015 API-650 ANSI/ASME B96.1 API standard 2000

6. Safety Case Strategy

Section 2 has described a classification scheme by which VHI and HI components of ABWR plant will be identified. VHI classification is assigned to components where certain postulated failure modes would result in intolerable radiological consequences and where it is impracticable to provide protection against failure. Typically, this applies to fault sequences associated with gross structural failure of a component, where the associated consequences include severe core damage and potential large uncontrolled releases. Accordingly, classification of VHI confers a requirement to justify a very high degree of structural reliability. For these components it is necessary to demonstrate that the likelihood of gross failure is so low that it can be discounted; this section proposes a strategy for a structural integrity safety case to substantiate such a claim.

The strategy outlined here accords with that expressed by TAGSI [Ref- 2] as their recommended approach for demonstration of ‘Incredibility of Failure’ in structural integrity safety cases. A four-legged format is adopted to provide diverse evidence of sound engineering practice so that the VHI components will remain serviceable throughout a specified plant life under all credible loading conditions, such that the possibility of disruptive failure without forewarning is sufficiently remote as to be deemed incredible. Within each of the four legs, a range of measures are identified that, in combination, are intended to evince the highest degree of structural reliability. The four legs of the safety argument are as follows:

> Leg 1, Interpolation/Extrapolation of Experience, establishes evidence of high integrity by good design and manufacture based on a proven track record. This provides a keystone for demonstration of high reliability and embodies the code and plant operating experience with objective of achieving quality of build, high integrity and the avoidance of defects.
>
> In Leg 2, Functional Testing, components are shown to be fit for purpose through effective functional testing such as pressure testing. It incorporates with operating experience embodied in design codes and provides some diversity and redundancy to pre-service inspection.
>
> Leg 3, Failure Analysis, supplements the Leg 1 in the aim of avoidance of defects to provide a safety case with both defect avoidance and defect tolerance, i.e that components are tolerant to through life degradation over the design life of the plant. It provides an assessment of through-life degradation mechanisms to demonstrate that in-service degradation will not threaten component integrity over a specific interval. This exceeds typical design code requirements by recognising that flaws may be present and establishing tolerance to them.
>
> Leg 4, Forewarning of Failure, gives contingency plan for the unexpected by providing diverse systems, such as in-service inspection and leak monitoring, to effectively forewarn of failure. This periodically confirms both the absence of unanticipated degradation and that known degradation mechanisms are not significant to nuclear safety.

This approach will be applied to VHI and HI components. The four legged safety arguments are intended to identify a suitable and sufficient diversity of evidence to demonstrate that VHI and HI components are sufficiently free of defects so that their safety functions will not be compromised at any time, that any identified defects are tolerable, and that the existence of defects that could compromise their safety function can be established through their life-cycle. For some HI components, parts of the above process may not be applied to the same level of rigour, nonetheless comprehensive evidence will be presented. For Class 1 components (and below), the multi-legged structure will not be followed, instead the safety case claim for these components will be based on the argument that the components have been designed and manufactured in accordance with recognised nuclear design codes, and suitable evidence provided to support this argument. This will in general be achieved by cross reference to component design reports.

For VHI and HI components, the safety case will identify a detailed and comprehensive description of the subject component, its duty and safety function(s) and boundaries in relation to other SSCs. Each leg of the safety case constitutes a broad claim, within each of which a number of subordinate arguments are identified, which in turn are supported by diverse sources of substantiating evidence. The details of the claims on which the safety case is to be founded will be developed during Step 2 of the GDA. The above describes the approach proposed for GDA but it is recognised that it may be developed further as the process of GDA advances.

7. Summary of Planned GDA Step 2 Structural Integrity Activities

In accordance with the discussion by Hitachi-GE and ONR, the following tasks are planned for Step 2 of the GDA.

Task a) Structural Integrity Classification Procedure

Building on the work started in GDA Step 1, a detailed methodology for establishing the structural integrity classification of the UK ABWR components will be developed. This document will describe the process for selecting components for more detailed assessment, describe the structure of the failure mode and effect criticality assessment (FMECA), describe how the direct and indirect consequences will be considered, describe the format and arrangements for the workshops including the required experience of the workshop panel members, describe how the components will be sub-divided for assessment and describe how the indirect effects on adjacent systems will be considered. The detailed procedure will focus on the RPV, Reactor Internals and piping systems inside containment. However, the process will be applicable to other systems outside containment. A selection of BOP components outside containment (such as the MSLs will be considered during Step 2). The report will identify the components and systems to be assessed during Step 2 of the GDA and will identify the plan for establishing the classification of the remaining locations. The report will be written to provide sufficient understanding of the proposed process to the ONR.

Task b) Weld Ranking Procedure

Hitachi-GE will prepare a document describing the procedure for the ranking of weld locations for subsequent defect tolerance assessment and NDE qualification. The procedure document will contain sufficient information to enable the ONR to understand the weld ranking procedure for the selected components. The procedure will be based on an assessment of inspectability and defect tolerance, and will identify the factors to be considered in the assessment, scoring criteria (e.g. stress, fatigue usage, residual stress, material, NDE access, geometry etc), weighting factors and a system for combining the inspectability and defect tolerance scores, along with other factors (such as a check to ensure that the selection of locations is suitably diverse and representative) to generate a prioritised list of assessment locations. The procedure will describe how the expert panels will be performed and the competence requirements of members. The procedure will be provided to the ONR for assessment.

Task c) Programme for defect tolerance assessment

Hitachi-GE will develop the detailed analysis plan for the assessment of selected weld regions of the RPV (initially), identifying what stress analyses will be required, what material properties will be used, what residual stress profiles will be used and how the R6 calculations will be performed. This will include a programme for the initial assessment locations on the RPV, and an outline plan for the subsequent assessments, noting that the full programme cannot be developed until the classification of all components has been established and the weld ranking completed. In developing the report, Hitachi-GE will establish the extent to which existing information can be used and whether additional stress analysis is required to support the R6 assessment.

Task d) Programme for Inspection Assessment

Hitachi-GE will describe how, in subsequent steps of the GDA, the effectiveness of the manufacturing NDE will be demonstrated for those components that have been prioritised from the ranking methodology. This will show how the effectiveness of the NDE will be demonstrated through a partial qualification process with evidence being presented through physical reasoning. It will also include the derivation of input information such as defect parameters and the roles and responsibilities of the different activities. The document will describe an initial programme for conducting the inspection assessments, starting with RPV welds noting that the full programme cannot be developed until the classification of all components has been established and the weld ranking completed.

Task e) Application of the Classification Procedure

Based on the procedure developed as part of Task a), a series of workshops will be held, involving experienced engineers from Hitachi-GE and the UK, to apply the proposed classification procedure to selected components. Separate workshops will take place for the RPV, Reactor internals, RIPs, MSLs inside containment, MSLs outside containment, other high energy pipework inside containment, selected low energy pipework and selected BOP components outside containment. On completion of the workshops, a report will be produced containing the FMECA tables and associated discussion to substantiate the classification of the selected components and to identify any further assessment requirements, such as impact analyses to substantiate indirect consequences. This report will be issued to the ONR for assessment.

Task f) Application of the Weld Ranking Procedure

Based on the procedure developed as part of Task b), Hitachi-GE will apply the weld ranking methodology to the components identified (based on the output of Task e) and will produce a prioritised list of locations for defect tolerance assessment and inspection qualification. The list will take account of the weld ranking, and will be reviewed to ensure that it contains a suitably representative set of weld type and components. The results will be included in a report which will be issued to the ONR for assessment.

Task g) Inspection qualification strategy

Hitachi-GE will produce an inspection qualification strategy that describes how the ENIQ methodology will be used to qualify the NDE systems performed at the end of manufacture for the VHI and HI components. It will describe the roles and responsibilities for the full range of qualification activities with particular emphasis on the formation of an independent qualification body. The strategy will describe how the inspection procedures and personnel will be qualified and the basis for conducting practical trials including test piece manufacture.

The strategy will also describe what measures will be taken to assure the reliability of the site inspections.

Task h) Step 2 Pre-Construction Safety Report (PCSR)

Hitachi-GE will develop a proposed structure for the Structural Integrity sub-chapter of the Step 2 PCSR, this structure will be provided to the ONR for comment. Based on this the sub-chapter will be developed, including the latest information based on tasks a) to g) above. A draft report will be provided to the ONR for review in advance of the formal issue of the document.

Hitachi-GE proposes a preliminary outline for Structural Integrity in the Step 2 PCSR as follows:

- Introduction

Presents the overall structure of the safety case to demonstrate that the class 1 SSCs can be assured over the 60 years design life to a level of structural reliability and a degree of rigour commensurate with the consequences of gross failure.

- Scope
  Describes the plant components that require substantiation with a series of Basis of Safety Case (BoSC) included as appendices.
- Objectives
  The objective of each BoSC is to support the claim that plant risk is tolerable and remains ALARP for the lifetime of the plant. Satisfying design basis for each SSC and the associated structural integrity safety functional requirements (SFRs) will support the claim.
- Safety Functional Requirements
  Presents the SFRs for each component to establish the specific role to maintain nuclear and radiological safety under normal and fault conditions. For example, the RPV will be required to maintain the highest reliability pressure boundary to contain the primary coolant, fuel etc. The SFRs for each component will be presented in the relevant BoSC.
- Structural Integrity (SI) Classification
  Classification Process: Describes the process by which each class SSC is classified
  Classifications: Discusses the SI Classification scheme, i.e. VHI, HI, Standard class 1
  Results of Classification Process: Discusses which SSCs as a result of the process have been allocated to each of the SI classifications.
- Basis of the component safety cases i.e. discusses the safety case structure for VHI, HI and Standard class 1 components and with reference to the appropriate appendices (e.g. RPV, MSL).
- Material
  Presents a basic plan to decide Material Specifications and Selection of Material Grade.
- Design Load Conditions
  Presents a basic plan to decide design conditions such as thermal cycles, pressure condition, external load, seismic load and so on to evaluate typical components.
- Conclusion
- References

The respective appendix for each component (e.g. RPV) will present the respective Basis of Safety Case (BoSC):

- Introduction
- Scope
  Presents arguments to support the claim that the nuclear and radiological risk for the component remains tolerably low for the design lifetime.
- Objectives
  Supports the claims made in the PCSR, i.e. risk for the component remains tolerably low and is ALARP for the design lifetime, substantiated by satisfying structural integrity safety design bases for all safety significant SSCs. SFRs are developed from the safety design bases and correspond to the functions that need to be maintained to provide assurance of nuclear and radiological safety.
- Interface with other safety case documents
- Description of plant component (e.g. RPV)
  Provides an overview of the plant component being discussed and provides references to more detailed reports.
- Safety Functional Requirements

Lists the SFRs for the component in question with postulated failure modes which result in the loss of the SFRs leading to identification of structural reliability targets commensurate with the consequences of gross failure.

- Classification of component
  Discusses assessment of component through consequences of failure, and outlines the result of the classification process with reference to more detailed information.
- Safety Case Claims, Arguments and Evidence (e.g. the four TAGSI legs for VHI components)
  The specific claims for the component in question will vary, but for the very high integrity components, the standard TAGSI approach is proposed (as discussed in section 6 of this report). For standard class 1 components, the claims will be based on demonstration of compliance with the relevant sections of the ASME BPVC.
  Arguments to support the safety case claims.
  Summary of evidence to support the arguments, including reference to the more detailed evidence reports
- Strengths, weaknesses and judgments in the safety case
  This is an important section and it is considered good practice that a safety case explicitly outlines the strengths, weaknesses and significant judgments of the proposal, rather than it being 'guess work' for the verifier /other stakeholders.
- Conclusion
- References

List of Documents for issue to the ONR during Step 2

The following will be prepared as part of the Step 2 programme.

- Structural Integrity Classification Procedure
- Weld Ranking Procedure
- Defect Tolerance Assessment Plan
- Inspection Assessment Plan
- Inspection Qualification Strategy
- Structural Integrity Classification Report
- Weld Ranking Report
- Proposed safety case structure for VHI and HI components
- Step 2 PCSR – Structural Integrity Sub-Chapter

The following will be prepared as the supporting documents for the Preliminary Safety Report on Structural Integrity (GA91-9901-0005-00001 Rev. C) for ONR's assessments.

- Summary of the Design of RPV for UK ABWR [Ref-7]
- Summary of the Design of Main Steam Piping for UK ABWR [Ref-8]
- Summary of the Design of Feedwater Piping for UK ABWR [Ref-9]
- Summary of the Design of Main Steam Isolation Valve for UK ABWR [Ref-10]
  Hitachi-GE provides the specific detail of the typical components like RPV, Main Steam Line Piping and Valves for UK ABWR in the area of Structural Integrity. These documents include an explanation of basic plan of material, manufacturing and inspections of typical components.
- Outline of the PSI and ISI Plan for UK ABWR [Ref-11]
- UK ABWR Approach for the Avoidance of SCC [Ref-12]

After GDA Step 2, Hitachi-GE will provide necessary information based on 'Generic Design Assessment Guidance to Requesting Parties' (ONR-GDA-GD-001 Revision 0) for Step 3 and Step 4 respectively.

GDA Step 3 is the process for overall design, safety case and security arguments review and its approximate timescale is twelve months. In GDA Step 3, Hitachi-GE will provide load conditions such as thermal cycles, pressure condition, external load, seismic load and so on for typical components to be evaluated.

GDA Step 4 is the process for detailed design, safety case and security evidence assessment and its approximate timescale is twenty eight months. In GDA Step 4, for typical components, Hitachi-GE will provide the final Fracture Mechanics Analysis reports, Stress Analysis and Evaluation report, Environmental Fatigue Evaluation report and so on.

8. Conclusion

This report outlines the strategy to be adopted by Hitachi-GE in justifying the structural integrity of components in the UK ABWR plant and has focussed on the strategy for identifying and justifying the VHI and HI components, of which it is intended to commence this work in GDA Step 2.

Section 2 describes a process of classification by which the structural integrity classification of UK ABWR components is to be established. For the VHI and HI components, key evidence in justifying their structural integrity will be provided by establishing avoidance of failure (Section 3), including materials selection and degradation issues and high reliability of inspections (Section 4).

A safety case strategy has been proposed for the UK ABWR plant in Section 6. For VHI and HI components, a four-legged safety case structure is proposed that is founded on evidence of high reliability by sound design, construction and functional testing, supplemented by failure analysis and provision to forewarn of failure. The diverse evidence necessary to substantiate the safety argument will be developed and implemented in each successive step of the GDA process.

9. References

[Ref-1] D A Miller and G Lawrence, Guidelines on Structural Integrity Related Safety Cases for Advanced Gas Cooled Reactors, Risk Assessment of Structures, IMechE Seminar, 10 December 1999.

[Ref-2] R Bullough, F M Burdekin, O J V Chapman, V R Green, D P G Lidbury, J N Swingler, R Wilson, The Demonstration of Incredibility of Failure in Structural Integrity Safety Cases, International Journal of Pressure Vessels and Piping 78, pages 539-552, 2001.

[Ref-3] R6 Assessment of the Integrity of Structures Containing Defects, Revision 4, EDF Energy.

[Ref-4] Approach to the adoption of Codes and Standards for the UK ABWR (GA91-9901-0008-00001, Revision A)

[Ref-5] Categorisation and Classification of Structures, Systems and Components (GA91-9901-0007-00001, Revision A)

[Ref-6] Initial safety case on Civil Engineering and External Hazards (GA91-9901-0004-00001, Revision A)

[Ref-7] Summary of the Design of RPV for UK ABWR (GA91-9201-0003-00035, Revision 0)

[Ref-8] Summary of the Design of Main Steam Piping for UK ABWR (GA91-9201-0003-00036, Revision 0)

[Ref-9] Summary of the Design of Feedwater Piping for UK ABWR (GA91-9201-0003-00037, Revision 0)

[Ref-10] Summary of the Design of Main Steam Isolation Valve for UK ABWR (GA91-9201-0003-00038, Revision 0)

[Ref-11] Outline of the PSI and ISI Plan for UK ABWR (GA 91-9201-0003-00039, Revision 0)

[Ref-12] UK ABWR Approach for the Avoidance of SCC (GA11-1001-0003-00001, Revision 0)

APPENDIX A – QUALITY CLASSIFICATION

A1 Introduction

Quality classification of nuclear power plant components is an important part of Japanese practice for the manufacture of such components. In this appendix, Japanese practice is outlined, together with a proposal to use the appropriate sub-sections of the ASME BPVC for quality classification of SSC in the UK ABWR.

A2 Japanese Practice

Table A1 illustrates the safety classification of plant SSCs in Japanese nuclear power plants.

Table A1 – JSME Classification

JSME Class	Major Definition	Major Equipments
Class 1, Core Support Structures	Reactor Pressure Boundaries Reactor Internals	RPV and connected piping to 2nd valves from RPV Reactor Internals
Class 2	Safety Systems	RCIC, HPCF, RHR, SLC, HCU, PCV Boundaries
Class 3	Safety Systems not defined as class 1 or 2	RCW, FPC, SFP, DG cooling pipe, R/W piping
Other	Pressure Equipment Regulation (JP)	Turbine

Japanese practice allocates quality classifications to the manufacture, construction, installation, commissioning and maintenance of plant components in accordance with their safety significance from JSME classification. This appears to be consistent with the UK ONR Safety Assessment Principles (SAPs), ECS.3, which state that ‘*Structures, systems and components that are important to safety should be designed, manufactured, constructed, installed, commissioned, quality assured, maintained, tested and inspected to the appropriate standards*’. Table A2 illustrates the definition of the Japanese quality classifications, A, B and C.

Table A2 – Quality Class Definition

Quality Class	Definition
A	(1) The reactor coolant pressure boundary, including isolation valves and mechanical supports. This class has the highest integrity, and the lowest provability of leakage. (2) Core support structure
B	(1) It limits the leakage of radioactive material from containment following a design basis accident. (2) The containment boundary including penetrations and isolation valves. (3) Reactor internal structure (4) It introduces emergency negative reactivity to make the reactor subcritical. (5) Directly engineering safety feature (6) It circulates a non-containment/non reactor coolant fluid to provide a post-accident safety-related function into and out of the containment.
C	(1) The other (not included in Class A and B) safety-related functions required to mitigate design basis accidents and other design basis event. (2) Indirectly engineering safety feature (3) Maintain spent fuel integrity, the failure of which could result in fuel damage such that significant quantities of radioactive material could be released from the fuel and result in offsite doses greater than normal limits. (4) The safety support function to Class A, B and C equipments such as, heat removal, room cooling, and electrical power. Emergency power supply (Diesel generator)

A3 Proposal for UK ABWR

For the UK ABWR, it is proposed that the appropriate ASME codes and standards will be applied for Class 1, 2 and 3 components, as shown in table A3 below.

Table A3 – Quality Class Definition for UK ABWR

Safety Classification	Quality Class	ASME III Sub Section	Major Equipments Systems
Class 1 (HR)	A	ASME III-NB +α	RPV (Highest Reliability parts)
Class 1 (Std.)	A	ASME III-NB ASME III-NG	RPV, RPV boundaries piping Reactor Internals (Need to discussion)
	B	ASME III-NG	Reactor Internals
		ASME III-NC	RCIC, HPCF, RHR, SLC
		ASME III-NE	RCCV
	C	ASME III-ND	RCW, FPC, DG
	D	ASME III-ND	Other in class 1
Class 2 (NSSS)	D	ASME III-ND	Class 2 NSSS

In table A4, a provisional allocation of the SSCs and appropriate ASME code standards is shown. This is subject to change via the formal classification process which will be implemented during Step 2 of GDA.

Table A4 – ASME Sec III sub-section requirements for UK ABWR

Safety Classification	Quality Class	ASME III Sub Section	Major Equipments Systems
Class 1 (HR)	A	ASME III-NB + α ASME III-NG + α	RPV (Highest Reliability parts)
Class 1 (Std.)	A	ASME III-NB ASME III-NG	RPV, RPV boundaries piping (Need to discussion)
	B	ASME III-NG	Reactor Internals
		ASME III-NC	RCIC, HPCF, RHR, SLC
		ASME III-NE	RCCV
	C	ASME III-ND	RCW, FPC, DG
	D	ASME III-ND	Other in class 1
Class 2 (NSSS)	D	ASME III-ND	Class 2 NSSS

Note that for VHI and HI components, the level of substantiation required is beyond ASME BPVC, and this is discussed in the main text.

To illustrate how the quality classification could in principle be applied to the UK ABWR, Figures A1 and A2 provide a representation of the main steam line and main feed water line respectively.

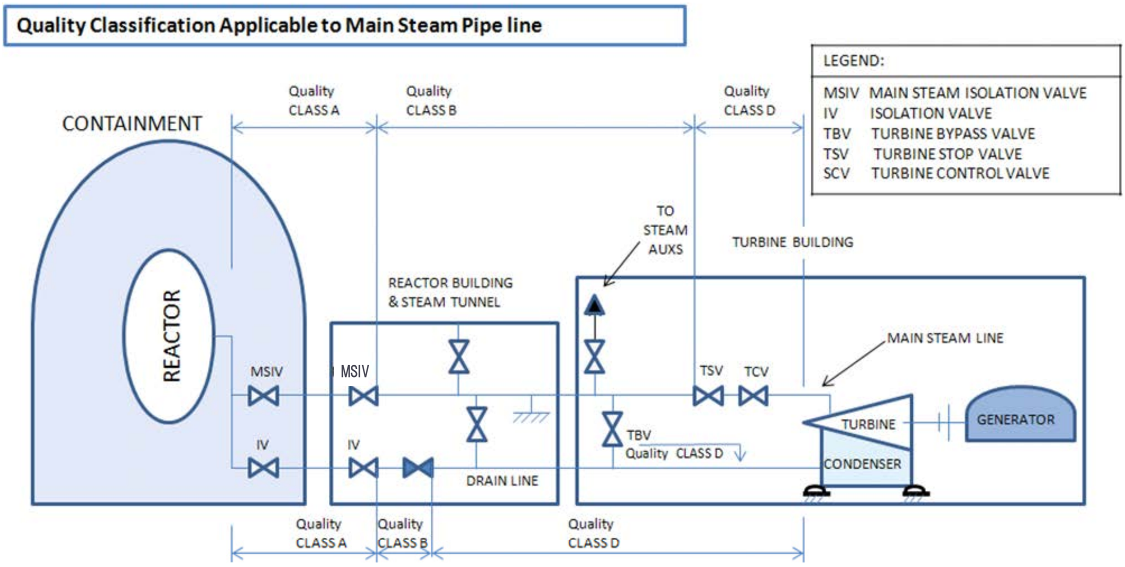

Figure A1 – Potential quality classification for main steam line

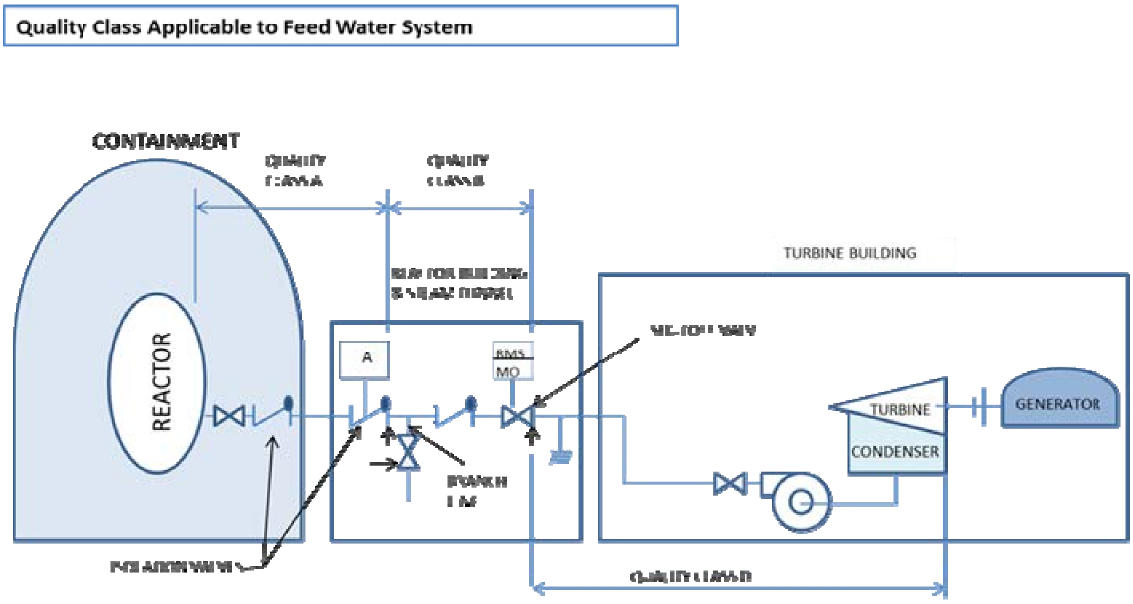

Figure A2 – Potential quality classification for main feed water line