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REVISION HISTORY

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01	Integration of co-applicant and INSA review comments	28.04.08
02	PCSR June 2009 update including: - Clarification of text, - Inclusion of references, - Erratum corrections, - New section 7 added: Comparison of requirements for break preclusion/non-breakable components with UK requirements for IOF.	29-06-2009
03	Consolidated Step 4 PCSR update: - Minor editorial changes - Update and addition of references - Introduction of High Integrity Component (HIC) safety case (§3) comprising §3.1 break preclusion piping requirements (previously §3), §3.2 non breakable component requirements (previously §6), and §3.3 comparison with IOF methodology (previously §7) - Deletion of secondary side chemistry requirements (previously §2); moved to new Sub-chapter 5.5 - Addition of information on the requirements for accessibility of welds (§5.3)	31-03-2011

Continued on next page

REVISION HISTORY (Cont'd)

Issue	Description	Date
04	Consolidated PCSR update: - References listed under each numbered section or sub-section heading numbered [Ref-1], [Ref-2], [Ref-3], etc - Minor editorial changes - Clarification of the scope of HIC in the introduction of section 3 and cross-reference added to the methodology presented in Sub-chapter 3.4 - Addition of new paragraph and reference in section 5.3 to present the analyses performed to ensure the accessibility and inspectability for ISI - Modification of section 5.5, "hydrostatic test" to be more consistent with UK regulation requirements	21-08-2012
05	Consolidated PCSR update: - Removal of cross-references to Sub-chapter 13.2 and update of text to refer to "HIC ... piping / components" (§3) - Inclusion of reference ENSDNR080245A regarding break preclusion requirements (§3.1.2.1.2, §3.1.2.2.2, §3.3.3) - Inclusion of text regarding 2A equivalent MCL section breaks (§3.1.2.2.2); deletion of §3.1.3 and §3.1.4 (concrete components, supports and structures). - New Table 7 "Assumptions additional to the break preclusion procedure on the main reactor coolant system pipework" added.	31-10-2012

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TABLE OF CONTENTS

SUB-CHAPTER 5.2 - INTEGRITY OF THE REACTOR COOLANT PRESSURE BOUNDARY (RCPB)

This sub-chapter describes how the integrity of the reactor coolant pressure boundary is ensured. The design rules and material specifications applicable to the reactor coolant pressure boundary are summarised in section 1. A description of the requirements applied to High Integrity Components is given in section 3. Section 4 describes the design criteria for the over-pressure protection system. An outline of the in-service inspection requirements is presented in section 5.

1. QUALITY LEVEL M1 REQUIREMENTS

1.1. DESIGN RULES FOR QUALITY LEVEL M1 EQUIPMENT

In accordance with Sub-chapter 3.2, pressurised components containing water directly heated by the nuclear fuel and which cannot be isolated from the reactor pressure vessel are subject to the provisions of the RCC-M (see Sub-chapter 3.8) applicable to class 1 equipment. This design rule is applicable to components whose failure would lead to a leak in excess of the make-up capacity of the chemical and volume control system (RCV [CVCS]).

In accordance with Sub-chapter 3.4, the design is based on the:

- Design conditions,
- Specified loads,
- Load combination rules.

This equipment is designed in accordance with RCC-M (see Sub-chapter 3.8), Volume B for diameters greater than or equal to nominal diameter 50 mm and Volume E below this nominal diameter.

In particular, Pressurised Thermal Shocks (PTS) are considered as part of the design justification for pressurised components.

1.2. MECHANICAL PROPERTIES OF THE COMPONENT MATERIALS OF QUALITY LEVEL M1 EQUIPMENT

The materials selected for the main components of quality level M1 equipment are generally those already in use for similar components on operational nuclear power plants, for which there is satisfactory operational feedback. However, other materials may be used provided appropriate justification is available.

The mechanical properties are defined in accordance with Volume I, Appendix ZI and Appendix ZIII of the RCC-M (see Sub-chapter 3.8) and consistently with the provisions of Volume II.

1.3. MATERIAL SPECIFICATIONS

The specifications applicable to materials used for parts subject to pressure from RCC-M class 1 reactor coolant system equipment (see Sub-chapter 3.8) are listed in chapter B 2000 of the RCC-M for existing materials, or in the equipment specifications for new materials.

The materials satisfy the applicable requirements of Volume II of the RCC-M. These material specifications may be viewed as standard for the listed applications.

2. REACTOR COOLANT SYSTEM WATER CHEMISTRY REQUIREMENTS

The primary side water chemistry is detailed in Sub-chapter 5.5.

3. HIGH INTEGRITY COMPONENT REQUIREMENTS

All RCPB main components and cooling piping (pressure boundary parts) are High Integrity Components (HIC) and can be listed in two categories originating from the EPR generic basic design:

- Non-breakable components: reactor pressure vessel, steam generator, pressuriser, reactor coolant pump
- Break preclusion piping: main coolant lines (excluding surge line and connected lines)

For the HIC break preclusion piping, the specific measures described in Sub-chapter 3.4 section 0.3 that are taken to demonstrate the integrity of HIC are detailed in section 3.1 and, for the specific case of demonstration of avoidance of fracture by propagation of crack-like defects the measures are described in section 1.6 of Sub-chapter 3.4.

For the HIC non-breakable components, the specific measures described in Sub-chapter 3.4 section 0.3 that are taken to demonstrate the integrity of HIC are detailed in section 3.2 and, for the specific case of demonstration of avoidance of fracture by propagation of crack-like defects the measures are described in section 1.6 of Sub-chapter 3.4.

A comparison between these requirements and the requirements conventionally applied to ‘Incredibility Of Failure Components’ (IOF components) in UK nuclear power plants is presented in section 3.3.

3.1. BREAK PRECLUSION PIPING REQUIREMENTS

3.1.1. Main requirements for the demonstration of Break Preclusion [Ref-1] [Ref-2]

This section describes the break preclusion principle as applied to the main RCP [RCS] pipework.

The approach to demonstrating break preclusion is based on the concept of multiple lines of defence-in-depth.

In order to guarantee the primary circuit pipework integrity throughout the plant life, and justify the claim that a pipe break is highly improbable, the following two lines of defence-in-depth are applied:

- Preventive measures based on design, verification of the design and manufacture/ manufacturing inspection, in order to make a failure highly improbable.

- Maintaining the system within its normal operating limits. This is achieved through the availability of protective devices (e.g. relief valves, etc.) and in-service surveillance (including in-service inspection) to detect any variation relative to normal operating conditions (e.g. loss of integrity).

The two lines of defence-in-depth listed above enable a guillotine break of the main reactor coolant pipework to be excluded from the design basis.

The following form two further lines of defence-in-depth:

- Limiting the consequences of failure,

- Controlling severe accidents.

As part of the two further lines of defence-in depth listed above, a break of connecting pipework is considered as a design basis accident.

Implementation of the break preclusion principle eliminates the need to design RCP [RCS] anti-whip devices, (used to limit the displacement of reactor coolant pipework) in the event of a guillotine break.

An overview of the break preclusion principle is illustrated in Sub-chapter 5.2 - Table 1.

3.1.1.1. Preventive measures

This section presents the measures taken under the first line of defence-in-depth to prevent any loss of integrity of the RCP [RCS] pipework.

The structural integrity of the RCP [RCS] pipework is based upon a design that reduces the likelihood of any damage occurring. The following types of damage to the RCP [RCS] pipework are considered:

- General damage, such as excessive/progressive deformation, or plastic instability, which may result from thinning of the wall through corrosion or general wear and tear

- Local damage or a pre-existing defect (or one initiated by fatigue or corrosion), which may result in a sudden leak or fast fracture risk, after a propagation phase

Preventive measures for potential damage are summarised in Sub-chapter 5.2 - Table 3.

3.1.1.1.1. *Prevention of general damage*

The prevention of damage from excessive deformation and plastic instability is ensured by control of the following three factors:

- The properties of component materials, elasticity limits, and tensile strength, at various operating temperatures
- The design basis that is a function of these properties and the specified loads and justification for the absence of significant wall thinning in operation
- Control of operational loads and limiting them to those encompassed by the design assumptions

Excessive deformation in itself does not constitute equipment failure, but may jeopardise margins relating to plastic instability.

Progressive deformation in itself does not constitute a means of equipment failure, but it may alter the verification conditions relating to the risk of plastic instability and fatigue, and may result in a redistribution of forces. In addition to the design features applied, prevention of this risk is also based on controlling loads under operational conditions.

Materials properties

The materials and manufacturing processes for the reactor coolant pipework are described in the RCP [RCS] section relative to reactor coolant pipework (see Sub-chapter 5.4). They are selected in accordance with:

- Homogeneity of properties within the wall
- No risk of significant creep within the range of operational temperatures
- Good weldability, guaranteeing equivalent properties in the material deposited and heat-affected zones
- Materials selected for the reactor coolant loops (wrought stainless steel and Narrow Gap Orbital TIG (NGOT) welding filler material) are not sensitive to thermal ageing given the expected ageing phenomena:
    - They demonstrate high toughness properties throughout the plant life
    - They are able to prevent any risk of fast fracture throughout the plant life, despite possible ductility loss
    - Ageing due to the effects of neutron irradiation, considered for the vessel, is negligible for main coolant pipework

Design basis

The RCC-M design and analysis rules define permissible methods and stresses guaranteeing sufficient margins relative to the risk of excessive deformation, particularly in relation to the minimum thicknesses required for design conditions. Protection from damage liable to jeopardise these margins is listed in the following:

- Protection from the risk of generalised corrosion is provided for reactor coolant system equipment by the selection of an austenitic stainless material or nickel-based alloys or by the welding of cladding on low alloy steel parts. The primary intention of these precautions is to limit corrosion products likely to be activated, or lead to load losses or reduction in heat transfer.

- Protection from the risk of erosion-cavitation is provided by the use of manufacturing experience in the design and layout.

- Erosion-corrosion risk depends on:

    - The fluid chemistry (pH, oxygen concentration) and temperature
    - Fluid speed, equipment geometry and the steam quality
    - The wall material (chromium, copper, and molybdenum content)
    - Duration of use

Eliminating the risk of erosion-corrosion under EPR operating conditions is achieved firstly by the choice of reactor coolant flow rates and chemistry, and secondly by the selection of materials (high chromium content stainless steels and nickel-based alloys).

Loads

The absence of fatigue damage throughout the plant life is based on full understanding of the operating transients (see Sub-chapter 3.4). Fatigue analyses demonstrate that in the absence of an initial flaw, the risk of cracks over the service period is extremely low (usage factor less than 1). Loads imposed on equipment in service are managed by control of operating parameters, pressure, and temperature. Overpressure protection mechanisms (see Sub-chapter 3.4) are also provided as part of the second line of defence-in-depth (see section 4 of this sub-chapter).

Other overall loads considered include external forces, thermal expansion, and direct or indirect hazards (see Sub-chapter 3.1). These forces are limited by the construction provisions selected and the usage precautions specified.

3.1.1.1.2. Prevention of local damage

Potential manufacturing defects

The risk of manufacturing defects existing in the equipment basic materials is considerably reduced because cast products are no longer used (all pipework components are forged). Parts produced by forging have a reduction ratio greater than three, which reduces the risk of defects, and contributes to giving the material homogeneous physical properties.

The risk of defects occurring during welding is considerably reduced by:

- A significant reduction in the number of welds: nine girth welds and integrated large branch connections (Surge Line, RIS [SIS], RRA [RHRS], RCV [CVCS]),

- Improving welding processes (NGOT) based on feedback,

- A significant reduction in the volume of the welds, which enables improved ultrasonic inspection, and therefore early detection of potential defect.

Initiation of potential defects in operation

Defects can be initiated during operation either by fatigue or through corrosion cracking.

Damage from cumulative fatigue and from progressive deformation is examined in the component's stress analysis file. The stress analysis file enables areas that are sensitive to such damage to be identified. In pressurised areas, for example, fatigue analysis must be carried out in three stages, namely analysis of the cracks initiation, crack propagation up to critical size and analysis of unstable cracks.

An area is considered to be at risk of cracking through fatigue or progressive deformation when:

- The usage factor is greater than 1
- The initiating factor is greater than 1
- The progressive deformation criterion is exceeded

Protection against fatigue damage is provided by taking precautions regarding:

- Vibration control (limited because of the high stiffness of the pipework)
- Control of low-cycle fatigue risk by design choices and load control (which enables usage factors lower than one to be ensured over the operating life)
- Reduction of the risk of thermal fatigue; three mixing zones are potentially affected these being the charging line nozzle, the pressuriser surge line nozzle on the hot leg and the RIS [SIS] Y-shaped nozzle on the cold leg. Design improvements have been made to these nozzles (in particular the removal of installation welds) which are now produced by forging. Furthermore, the loads experienced by the RCV [CVCS] pressuriser surge line and RIS [SIS]/RRA [RHRS] Y-shaped nozzles are reduced because of improvements in reactor operation (pressuriser regulation, duplication, and increase in diameter of the charging line, RRA [RHRS] connection at 120°C).
- Reduction in the risk of stratification in the surge line through design choices, operating conditions and procedures. (The surge line itself is not part of the « break preclusion » zone).

Local corrosion phenomena are avoided by carefully selecting suitable materials for the chemistry of their operating environment:

- The absence of intergranular corrosion of austenitic steels is ensured by the use of very low carbon steels or titanium or niobium stabilised steels. Low carbon steel was chosen for the main RCP [RCS] pipework, which does not need to be stabilised.
- The risk of stress corrosion is associated with the combination of three factors:
    - The environment
    - High temperatures
    - High levels of permanent stress

This risk is also avoided by the use of materials that are not susceptible to this type of corrosion in the primary environment (NC 30 Fe alloy widely used instead of NC 15 Fe alloy [Ref-1]) the test results for which demonstrate that the new grade used is not liable to stress corrosion in reactor coolant.

- Pitting corrosion is likely to affect only the external surface of the main reactor coolant system and steam lines
- As regards the primary lines, and in particular their dissimilar welds, the corrosion risk is associated with potential leaks, notably during phases of refuelling

Loads

Loads may cause initiation of defects, defect propagation, and possibly the risk of fast fracture. Protection from this risk is based on the understanding and surveillance of operational situations, in the same way as is applicable to general damage.

Properties of materials

The materials selected for the reactor coolant loops (forged stainless steel and welds NGOT) are not sensitive to thermal ageing and have high toughness properties, giving excellent reduction in the risk of fast fracture throughout plant life.

The risk of brittle fracture is excluded for reactor coolant pipework, as it is made from austenitic stainless steel. The absence of any risk of ductile break is also demonstrated.

Tolerance to large defects

The reactor coolant pipework is tolerant to large defects due to the demonstration that fast fracture risks can be excluded from the design basis.

Risks of indirect damage

This involves, for example, the risk due to objects falling on the reactor coolant pipework, risks due to a pipe break in earthquake conditions, and risks due to guillotine pipe break in the case of no break preclusion.

3.1.1.2. Surveillance measures

This section presents the measures taken under the second line of defence-in-depth to prevent any loss of integrity of the RCP [RCS] pipework.

3.1.1.2.1. Operational surveillance

- A system of recording transients will be implemented in accordance with the existing practices for reactors operated in France and Germany:
  - Temperature and pressure are monitored for transients in normal, abnormal, and exceptional situations
  - The relevant transients are compared with design basis transients

  - o Surveillance measures ensure that the actual number of design basis transients does not exceed the number of allowable occurrences. Any variation compared to the design basis transients list will be evaluated in terms of damage

- The RCP [RCS] water chemistry parameters are verified during operation of the plant and during shutdowns (boron, hydrogen, pH, etc.)

- Appropriate instrumentation will be put in place on the lead unit in order to improve assessment of thermal loads, monitor local thermal-hydraulic performance in the mixing zones and, if necessary, the overall performance of the systems

3.1.1.2.2. In-service inspection of main reactor coolant pipework

In-service inspection is intended to confirm the absence of notable defects in a number of locations selected in advance. This is one of the components of in-service surveillance for the second line of defence-in-depth in the demonstration of break preclusion, and is therefore independent of the design and manufacturing quality process.

The detailed programme will be finalised during the detailed design phase and will take into account feedback experience, available non-destructive testing (NDT) techniques, the recommendations of the Safety Authority, and the results of Pre-Service Inspection (PSI) before start-up.

The in-service inspection programme is a surveillance programme optimised by the break preclusion principle by permitting the inspection of areas that would not be accessible if anti-whip mechanisms were installed.

All welds, and more generally all areas of the reactor coolant system loops subject to an in-service inspection programme, will be accessible and be inspected in accordance with the technical guideline requirements.

In such areas, the non-destructive testing equipment used will be qualified. All homogenous welds will allow inspection by at least one volume inspection method. Heterogeneous welds will allow inspection by two volume inspection methods in accordance with the technical guideline requirements.

3.1.2. Requirements for further lines of defence-in-depth

The purpose of the following two further lines of defence-in-depth (independent of the previous lines of defence-in-depth) is to prevent and limit the consequences of accidents:

- The first line involves the prevention and limitation of accidents postulated in the reference design basis for the plant (leak detection, LOCA, migration)

- The second consists of dealing with accidents that are not postulated within the reference design basis (2A LOCA)

3.1.2.1. First further line of defence-in-depth

3.1.2.1.1. Tolerance to large through -wall defects

Tolerance to large through-wall defects is achieved if a significant through-wall defect is stable and would result in a detectable leak rate before a 2A break of the pipework. It provides an additional line of defence-in-depth to the break preclusion principle, for limiting the consequences of a loss of integrity.

To this end, it is demonstrated that for all girth welds on the reactor coolant pipework there is a sufficient margin between:

- The smallest through-wall crack size for which leakage at the break is detectable, and
- The largest critical through-wall crack size

3.1.2.1.2. Leak detection via the main reactor coolant system

Leak detection function requirements [Ref-1]

Leak detection contributes to the prevention of reactor coolant system loop breaks by detecting any through-wall cracks that may appear in service before they reach a critical size.

The safety function of the leak detection system is consequently similar to a line of defence associated with limiting the consequences of a loss of integrity. Leak detection will initiate alarms in the event that the flow rate exceeds certain defined limits.

Leak detection systems

Leaks inside the containment are detected and located by equipment positioned within the cells, ventilation system, tanks, and blowdown/vent path.

To detect a leak within the containment, a combination of some of the methods listed below is used (this is not an exhaustive list):

- Moisture sensors (based on the dew point); these are sensitive devices but they provide no data on the size of the leak
- Temperature sensors; these are sensitive devices but they provide no data on the size of the leak
- Condensate flow meters positioned on the containment cold ventilation coil. Condensate is collected in a chamber. Data supplied by the sensors is accurate for condensates and may indicate the rate of steam condensation
- Radiation monitors on the containment enclosure sampling system; these are sensitive devices but they provide no data on the size of the leak
- Water level measurement in the blowdown, venting and exhaust chambers or sumps; these are sensitive devices and provide some data on the size of the leak

In addition, radioactive leaks such as leaks of reactor coolant to the secondary side are detected and located through N16 radiation measurements.

3.1.2.2. Second further line of defence-in-depth

3.1.2.2.1. Mechanical design of heavy components

The mechanical design of supports and heavy components is based on breaks in pipework connected to the reactor coolant system, the most significant being a break in the pressuriser surge line.

From an engineering perspective, this means that:

- The postulated break will not result in failure in the reactor coolant pipework, nor in a complete break in any connected lines
- The mechanical strength of the RPV internals will make it possible to ensure core cooling
- The geometry of the rod cluster control assembly (RCCA) guide tubes will enable rod drop and core shutdown
- The integrity of component supports is verified
- The integrity of reactor coolant pumps, steam generators (see Sub-chapter 5.4) and of their internal systems is verified

Postulated pipe breaks are detailed in Sub-chapter 5.2 - Table 2.

However, to reinforce the stability of large components (vessel, reactor coolant pumps, steam generators), a static load is taken into account for the design basis of their supports, i.e. 2pA, where p is the operating pressure (155 bar) and A is the pipework cross-section area.

No mechanical damage is transferred to unbroken loops. The reactor vessel is a fixed point standing on a resistant support that limits loop stresses. Each loop is surrounded by reinforced concrete walls that are sufficiently thick to prevent damage to unbroken loops. The walls are designed for the loads indicated in Sub-chapter 5.2 - Table 2.

3.1.2.2.2. Design basis accidents

The safety objective of these analyses is to demonstrate:

- The sub-criticality of the core by the reactor trip system
- The attainment of a long-term safe shutdown state
- Cooling of the core
- Non-escalation of events

The most significant break postulated as a design basis accident is a complete break in the pressuriser surge line. This is analysed in the PCSR with the existing worst-case rules and assumptions that are used in design basis accident analysis (see Chapter 14).

However, a 2A equivalent section break in the main coolant lines must also be considered as part of the additional measures. The objective is to reduce the severity of beyond design basis accidents, especially those that could occur despite the implementation of the Break Preclusion concept.

This analysis must allow the definition, using realistic assumptions, [Ref-1] of:

- the Safety Injection System (RIS [SIS]) capacity,
- the resulting pressure and temperature and the verification of containment building resistance,
- the post accident conditions to be used for equipment qualification.

These further defence in depth assumptions are summarised in Sub-chapter 5.2 - Table 7.

3.2. "NON BREAKABLE" COMPONENT REQUIREMENTS

3.2.1. Special requirements

The following section specifies requirements for the design, manufacture, inspection and in-service surveillance of nuclear pressurised equipment in the basic nuclear installation that are classified as “non breakable”. The requirements also apply to the secondary side of the steam generators.

The failure of a class 1 pressurised equipment that may lead to situations for which the safety report does not provide any measures to recover a safe state are known as "non breakable". High standards of quality assurance throughout all stages of design, procurement, manufacture, installation, and inspection are applied in accordance with level 1 RCC-M requirements. Such equipment is subject to special specifications with additional requirements and has special in-service surveillance.

Firstly, the most demanding situations to which the equipment may be subjected are accurately evaluated taking experience and system study results into account, covering normal operation, plant transients, faults, and internal and external hazards. These circumstances are subject to regulatory monitoring by the review of reactor operations, and, if needed, using additional special instrumentation in those areas subject to local loading (such as mixing areas, sensitive areas, etc.).

Secondly, all potential causes of damage are subject to an explicit evaluation at the design stage (stress reports or behaviour analysis file), taking account changes in material properties under operating conditions. These changes are liable to be subject to a surveillance programme in case they become significant (e.g. the irradiation surveillance programme for the vessel core shells).

Thirdly, component materials are selected based on significant experience of their proper performance under manufacture and in service. Specified properties must conform to regulatory requirements for class 1 equipment that requires the highest level of manufacturing quality. Materials and parts are subject to technical qualification at the procurement stage as soon as any risk of variation in their properties (related to development or the complexity of the planned manufacturing process) is identified. The processes used to manufacture forged components provide sufficient reduction and, if required, a suitable inspection of inclusions is performed after manufacture.

All manufacturing operations are subject to technical qualification. The purpose of this is to ensure that components manufactured under the conditions and according to these qualifications will have the required properties. Welding procedures, including cladding by welding, and the staff carrying out those procedures are qualified according to strict rules and approved by an authorised body.

Non-destructive tests, carried out by qualified staff with the appropriate skill levels and approved by a recognised third-party body, allow unacceptable manufacturing defects to be detected. Pressure-resistant permanent assemblies are all subject to a full inspection and an inspection of all final surfaces on components is carried out.

An operational inspection programme is implemented to detect any risk of loss of integrity that might occur despite the precautions taken at the design and production stages, and the monitoring of operating conditions. The programme provides the necessary resources to determine the nature, origin, and possible development of defects and damage observed on equipment. Protection systems are also provided to avoid exceeding the operational conditions. Defects and damage are repaired if there is convincing proof that this will not result in equipment failure. The non-destructive tests are performed by qualified staff who have the appropriate skills and are approved by a safety body authorised for the purpose under the maintenance and monitoring operations programme. The non-destructive test procedures used are subject, prior to their use, to approval by an accredited body, with the aim of demonstrating that the procedures achieve their stated objectives.

A process for review of experience from others facilities has been used to design the EPR, which includes the understanding of degradation mechanism. Potential in-service degradation mechanisms are consequently considered.

3.2.2. Protection against Internal and External Hazards

In addition to the special requirements applied to the pressurised equipments, the whole Reactor Coolant System (RCP [RCS]) is subject to protection against internal and external hazards.

The following external hazards have been considered from the point of view of their effects upon the RCP [RCS] lines and are described in Sub-chapter 13.1:

- Seismic event
- Aircraft crash
- External explosion
- Lightning and magnetic interferences
- Underground water
- Extreme meteorological conditions (temperature, snow, wind and rain)
- External flooding
- Offsite hazardous substance

The reactor building protects the RCP [RCS] against most of these external hazards. The reactor building cooling system protects the RCP [RCS] from extreme ambient temperature. The reactor building and the RCP [RCS] have been assigned seismic category 1.

The RCP [RCS] is designed to maintain structural integrity during a Safe Shutdown Earthquake (SSE) event.

The following internal hazards have been considered from the point of view of their potential effect on the RCP [RCS]:

- Fire
- Missiles
- Failure of pressurised components
- Main turbine disintegration
- Dropped loads
- Explosive gas mixtures
- Hazardous materials
- Explosive effects of electrical faults
- Radio-frequency interference
- Flooding

The primary system pipework is located inside bunkers that protect it from missiles arising inside the containment. As it is all located inside the containment, it is protected from missiles arising in the auxiliary building.

The design of the polar crane, which is not operational whilst the plant is at pressure, limits the probability of dropped loads.

3.3. COMPARISON OF REQUIREMENTS FOR BREAK PRECLUSION / NON BREAKABLE COMPONENTS WITH UK REQUIREMENTS FOR IOF COMPONENTS

3.3.1. Introduction

This section reviews the design, manufacturing and operational requirements placed on Break Preclusion (BP) and non-breakable components, and compares them with UK requirements for components for which gross failure is outside the design basis defined in the nuclear safety case. Such components are generally referred to as Incredibility of Failure (IOF) components in the UK context.

To make a comparison with UK IOF requirements, a definition of IOF is first provided. Comparisons are then made between the requirements for BP/non-breakable components and typical requirements for IOF components.

3.3.2. Requirements for UK safety cases for IOF

HSE Safety Assessment Principles (SAPs) [Ref-1] relating to the integrity of metal components and structures identify a class of components which form “the principal means of ensuring nuclear safety” and for which the estimated likelihood of gross failure needs to be very low to ensure that the risk of an unacceptable off-site radiological release is tolerable. For these components, arguments must be provided that gross failure is so unlikely that the consequences need not be considered in the safety case for the facility. Such components are frequently referred to as “Incredibility of Failure” (IOF) components. HSE SAPs for structural integrity, and the applicable supporting Technical Assessment Guide, T/AST/016 [Ref-2], require that for this type of component, structured arguments and evidence are provided to support the claim that gross failure is so unlikely that it can be discounted.

The ONR expectation of an IOF safety case is that it be structured around the multi-legged / multi-element concept proposed in by the UK Technical Advisory Group on Structural Integrity (TAGSI) [Ref-3]. It was considered useful to compare the EPR approach to discounting gross failure with the TAGSI approach for demonstrating IOF.

The TAGSI approach to demonstrating IOF is based on providing arguments in four independent “Legs”, each of which consists of number of discrete elements. The multi-legged / multi-element structure can be represented graphically, as illustrated in Sub-chapter 5.2 - Table 4. The TAGSI Legs are listed below:

- Leg 1 - Design and Manufacture
- Leg 2 - Functional Testing
- Leg 3 - Failure Analysis
- Leg 4 - Forewarning of Failure

The TAGSI approach to an IOF safety case seeks to provide defence-in-depth by provision of multiple arguments which are, as far as possible, independent so that weaknesses in one argument are compensated for by strengths in the others.

The TAGSI approach states that there is no unique leg / element structure for an IOF safety case; the important point is not that the legs should be completely independent, but that together they should form an adequate level of confidence of reaching a low target failure frequency. The probabilistic target for an IOF safety case, defined in the HSE SAPs and by the TAGSI, is a failure frequency equal to or less than 10^{-7} per year.

3.3.3. Comparison of the Break Preclusion principle with the TAGSI approach for IOF

For EPR, the BP principle [Ref-1] is invoked for pipework for which the failure frequency is so low that the consequences of gross failure need not be considered in the safety case. Section 3.1 of this sub-chapter describes the application of the BP principle to major RCP [RCS] pipework, and Sub-chapter 10.5 describes the application to the main steam lines inside and outside the containment. The following sub-sections compare the BP requirements with the TAGSI approach to demonstrating IOF.

3.3.3.1. Comparison of multi-legged / multi-element approach and BP approach

The BP requirements are summarised in Sub-chapter 5.2 - Table 1, which shows that the demonstration of BP is based on the concept of multiple lines of defence-in-depth. Four lines of defence are identified as follows:

- Damage prevention by good quality design and manufacture
- Operational surveillance. This line of defence includes operational monitoring and in-service inspection.
- Mitigation. This line of defence includes measures to prevent failure escalation. It includes measures to prevent design basis faults escalating to cause gross failure of BP components, analysis to confirm of tolerance to through-wall defects, measures to detect leak before break, etc.
- Risk reduction. This line of defence is applied to major primary and secondary coolant pipework subject to the BP principle. It involves making design provisions to ensure that the consequences of gross failure will not lead directly to severe core damage or an unacceptable release of radioactivity outside the reactor containment.

Sub-chapter 5.2 - Table 5 compares the specific requirements imposed to achieve the above four lines of defence with the elements identified in the four ‘Legs’ of the UK TAGSI approach. It is seen that all the TAGSI elements are covered in the first three lines of defence in the BP approach. The fourth line of defence in the BP approach, i.e. design measures to prevent unacceptable radiological consequences of gross failure, is not required in the TAGSI approach due to the implicit assumption that gross failure of IOF components would lead automatically to unacceptable off-site radiological consequences.

3.3.3.2. Independence of the arguments in the BP approach

The TAGSI approach is to show that the legs of an IOF safety case are to some degree independent, and that together they give confidence that a failure frequency of 10^{-7} has been reached. The BP principle is implemented by applying successive lines of defence-in-depth, which are also independent, and which together are sufficient to enable gross failure of a Break Preclusion component to be discounted. The fourth line of defence-in-depth in the BP approach (Risk Reduction) provides an additional independent level of protection against failure consequences which is beyond the protection required in the TAGSI approach.

3.3.4. Comparison of requirements for non-breakable components with the TAGSI approach for IOF

EPR pressurised components in the primary and secondary circuits are classified as “non-breakable” if their failure may lead to situations where no measures are available to recover a safe state. The special requirements for non-breakable components are described in section 3.2 of this sub-chapter. These measures include:

- Use of high standards of quality assurance applied in design, procurement, manufacture, installation, and inspection in accordance with Level 1 RCC-M requirements [Ref-1].

- Confirmation of integrity of components in loading conditions for all circumstances, including normal operation, plant transients, faults, and internal and external hazards. Use of additional special instrumentation where appropriate in sensitive areas or areas subject to localised loading.

- Use of surveillance programmes to monitor changes to material properties over component life.

- Requirements for component materials properties to conform to regulatory requirements for Level 1 RCC-M appropriate to highest level of manufacturing quality. Use of forged manufacturing techniques where practicable and manufacturing inspections to ensures low probability of defectiveness.

- Manufacturing operations subject to technical qualification to ensure required quality standards. Welding operations carried out by qualified staff according to strict rules approved by an authorised body.

- Non-destructive tests, conducted by qualified staff approved by a recognised third-party body, carried out to detect manufacturing defects and detect and monitor defects during operation.

- Use of feedback experience on in-service degradation mechanisms from other facilities in component design.

Sub-chapter 5.2 - Table 6 compares the requirements for non-breakable components with the elements identified in the four ‘Legs’ of the UK TAGSI approach. It is seen that, as for BP components, all the TAGSI elements are covered by the requirements for non-breakable components.

As for BP components, successive independent lines of defence-in-depth can be identified, which are equivalent to the independent argument legs of the TAGSI approach.

Although gross failure of non-breakable components is not considered in the design of EPR protection systems, the off-site radiological risk due to failure of non-breakable components is included in the PSA analysis for the reactor presented in Chapter 15 of the PCSR. Due to the capability of the containment building to withstand the severe accident conditions that could result from the failure of the non-breakable components, the risk from such failures is assessed as negligible.

4. OVER-PRESSURE PROTECTION OF THE REACTOR COOLANT SYSTEM REQUIREMENTS

Over-pressure protection is intended to protect the integrity of the main reactor coolant system under hot and cold conditions. It is achieved by the use of pressure relief valves in parallel with the reactor protection system and related equipment.

This topic is covered in Sub-chapter 3.4 section 1.5.

4.1. DESIGN BASIS

Over-pressure protection of the RCP [RCS] under hot and cold conditions is achieved by three protection lines connected to the pressuriser. The pressure relief valves fitted on these lines are described in Sub-chapter 5.4.

Discharge from these pressuriser pressure relief valves is provided by the pressuriser discharge system described in Sub-chapter 5.4. The relief valves provide protection from over-pressure in hot or cold conditions. Cold opening of the pressuriser relief valve occurs following a dedicated initiating signal at a certain reactor coolant system pressure setpoint. This signal acts directly on the relief valve controlling actuator. During hot RCP [RCS] over-pressure transients of category 3 or 4, each pressuriser relief valve is automatically activated when the pressure reaches the setpoint of the actuator that controls it.

The overpressure protection mechanisms are designed and defined to meet the general requirements defined for each accident category.

For hot and cold conditions, transients have been studied by selecting the most conservative assumptions for the initial design basis parameters for the over-pressure protection system (see relevant paragraphs below).

4.1.1. General requirements

The general requirements are given in Sub-chapter 3.4 section 1.5.

4.1.2. Over-pressure protection at power

Protection against over-pressure at power in the reactor coolant system (and the secondary cooling system) is detailed in Sub-chapter 3.4 section 1.5.

4.1.3. Over-pressure protection in cold shutdown state

In addition to the general requirements, reactor coolant pressure must not exceed the allowable pressure-temperature limits for the reactor vessel addressed in Sub-chapter 5.3.

The setpoint for opening the pressuriser relief valves is automatically adjusted depending on the state of the plant. The definition of the setpoint value in cold conditions serves to limit the maximum pressure reached in the reactor coolant system.

The cooling system in the shutdown state RIS/RRA [SIS/RHRS] is protected against over-pressure by the pressure relief valves that are specific to this system. Over-pressure protection for the RRA [RHRS] is studied in Sub-chapter 6.3.

Cold over-pressure protection for the reactor coolant system is detailed in Sub-chapter 3.4 section 1.5..

4.2. DESIGN EVALUATION

The discharge capacities of the pressuriser relief valves and steam generators safety valves are determined from the postulated over-pressure transient conditions in conjunction with action by the reactor protection system. An evaluation of the system’s functional design and analyses of the system's ability to fulfil its functions have been carried out.

4.3. PIPING AND INSTRUMENTATION DIAGRAMS

Over-pressure protection for the RCP [RCS] is provided by the pressuriser relief valves. Flow diagrams for the RCP [RCS] system are to be found in Sub-chapter 5.1.

The pressuriser discharge system is described in Sub-chapter 5.4.

The steam generator safety valves are described in Sub-chapter 10.3.

4.4. DESCRIPTION OF EQUIPMENT AND COMPONENTS

The operation, significant design parameters, quantity, operating cycles, and environmental conditions for the pressuriser relief valves are addressed in Sub-chapter 5.4.

Components of the protection system against steam system over-pressure are presented in Sub-chapter 10.3.

4.5. ASSEMBLY

The assembly of the components of the reactor coolant system over-pressure protection system is addressed in Sub-chapter 5.4.

The assembly of the components of the steam system over-pressure protection system is addressed in Sub-chapter 10.3.

4.6. APPLICABLE CODE AND CLASSIFICATION

The safety classification of the reactor coolant system over-pressure protection system is given in Sub-chapter 3.2.

4.7. MATERIALS SPECIFICATIONS

The materials for the reactor coolant system and its fastenings are specified by the RCC-M (see Sub-chapter 3.8).

4.8. INSTRUMENTATION

Each pressuriser relief valve is fitted with appropriate temperature measurement instrumentation to warn the operator of a steam discharge due to a leak or operation of the valve.

Hydrogen build-up, upstream of the relief valves, can also be detected by temperature measurement in the relief valve inlet pipework.

An indication of the status of each relief valve is displayed in the control room.

4.9. SYSTEM RELIABILITY

The reliability of the pressure relief mechanisms has been demonstrated through testing and operational experience; the qualification of the relief valves is addressed in Sub-chapter 5.4.

4.10. TESTING AND INSPECTION

The testing of the components of the reactor coolant system over-pressure protection system (primarily the relief valves) is addressed in Sub-chapter 5.4.

The design of the relief valves and related components allows their periodic dismantling and inspection. All valve parts (except for welded connections) can be dismantled for inspection and replacement. Those parts that are not directly removable belong to sub-assemblies that can be inspected in the workshop and are interchangeable.

5. IN-SERVICE INSPECTION OF THE REACTOR COOLANT SYSTEM

This section concerns the in-service inspections that are carried out in general on the Reactor Coolant System as part of ensuring the integrity of its pressure boundary. It also applies to the secondary side of the steam generators.

5.1. MAIN POTENTIAL DAMAGE TO BE TAKEN INTO ACCOUNT

The potential causes of damage are the same as those detailed in section 3.1.1 of this sub-chapter.

5.2. RCPB IN-SERVICE INSPECTION PROGRAMME

All class 1 mechanical components, such as the reactor pressure vessel, the main coolant lines (including the surge line), the steam generators and the pressuriser, which require in-service inspections, are designed, manufactured and assembled to permit all welds and all areas to be inspected.

The following table establishes an initial provisional list of the typical areas that could be subject to in-service inspection. This analysis is based both on the experience gained from similar designs and on specific analysis carried out for the EPR.

Component	Sensitive area	Damage
Reactor Pressure Vessel	Stud and threaded holes	Fatigue
Main Coolant Lines	RCV [CVCS] charging nozzles	Fatigue
	RIS [SIS] inclined nozzles	Fatigue
	Surge Line nozzle	Fatigue
Steam Generator	Tube sheet / channel head weld	Fast fracture
	Tube sheet / secondary shell weld	Fast fracture
	Channel head / partition plate weld	Fatigue
Pressuriser	Heater sleeve weld	Fatigue

The in-service inspection programme has been based on the mechanical analysis results (fatigue, fast fracture, etc) and on feedback knowledge in specific areas (mechanical problems, for example). It has been verified that there are no accessibility problems for any area (see section 5.3). In addition, it has been verified that the expected performances during the non-destructive in-service inspection tests are consistent with design and manufacture (surface quality, geometry, etc.). Non-destructive tests during manufacture must show that no unacceptable defects exist. For all workshop welds that require in-service inspection, non-destructive tests are carried out using the same method. They have the status of "preliminary inspections" which provide a reference point for subsequent inspections.

Detailed analyses identify the areas that are potentially sensitive to damage such as fatigue or fast fracture. In this case, these areas are included in the in-service inspection programme.

A selection of welds where the combination of loads and materials properties is the most unfavourable, along with a selection of less sensitive welds, are included within the in-service inspection programme as part of defence-in-depth. The in-service inspection programme is reduced compared to the equivalent programme for sensitive areas if it has been confirmed during the analysis:

- There is no risk of damage during operation
- The design fulfils the RCC-M criteria (see Sub-chapter 3.8)
- During manufacture, and based on non-destructive tests, there are no unacceptable defects in these areas

The non-destructive tests are qualified in accordance with relative regulatory rules.

5.3. ACCESS NECESSARY TO INSPECT THE RCPB

Welds forming part of the RCPB must be accessible and able to be inspected.

The UK EPR design takes into account the requirement to access the areas to be controlled during the pre-service inspection and the in-service inspections; for each inspection area, there is adequate design for accessibility and controllability of the welds to be inspected [Ref-1].

In particular a comprehensive study of the end-of-manufacturing accessibility and inspectability of the Main Coolant Line (MCL) welds has been performed [Ref-2], especially for relevant UT techniques. This study covers the MCL design state after modification of the crossover leg as described in Sub-chapter 5.4 section 3. It has also covered in-service accessibility and shows that this accessibility is sufficient to enable the deployment of the Pre-Service / In-Service Inspections yet to be developed.

Pipework systems that require surface or visual inspection are designed to enable suitable access and visibility to allow such inspections to be carried out properly. Access for in-service inspections of the key components of the reactor coolant system other than the reactor vessel will be provided in the following way:

- In general, work platforms or temporary scaffolding will be supplied to facilitate access to the areas to be inspected
- Manholes are designed for entry into the steam generator water chamber to provide access for internal inspection
- A manhole is built in to the upper spherical head of the pressuriser to permit internal inspection
- The insulation covering all component welds and the adjoining pipework is removable in those areas where external inspection is planned
- The reactor pit is designed with an access area reserved for staff during refuelling operations to permit external inspection of pipework and heavy components

5.4. INSPECTION TECHNIQUES AND PROCEDURES

Various examination equipment, procedures and techniques are available to carry out in-service inspection. The specific inspection techniques and procedures and the inspection tools used will be selected before the in-service inspection period.

At present, the methods that can be used are as follows:

- Radiographic inspection
- Ultrasonic inspection
- Dye penetrant testing
- Magnetic particle examination
- Eddy current inspection

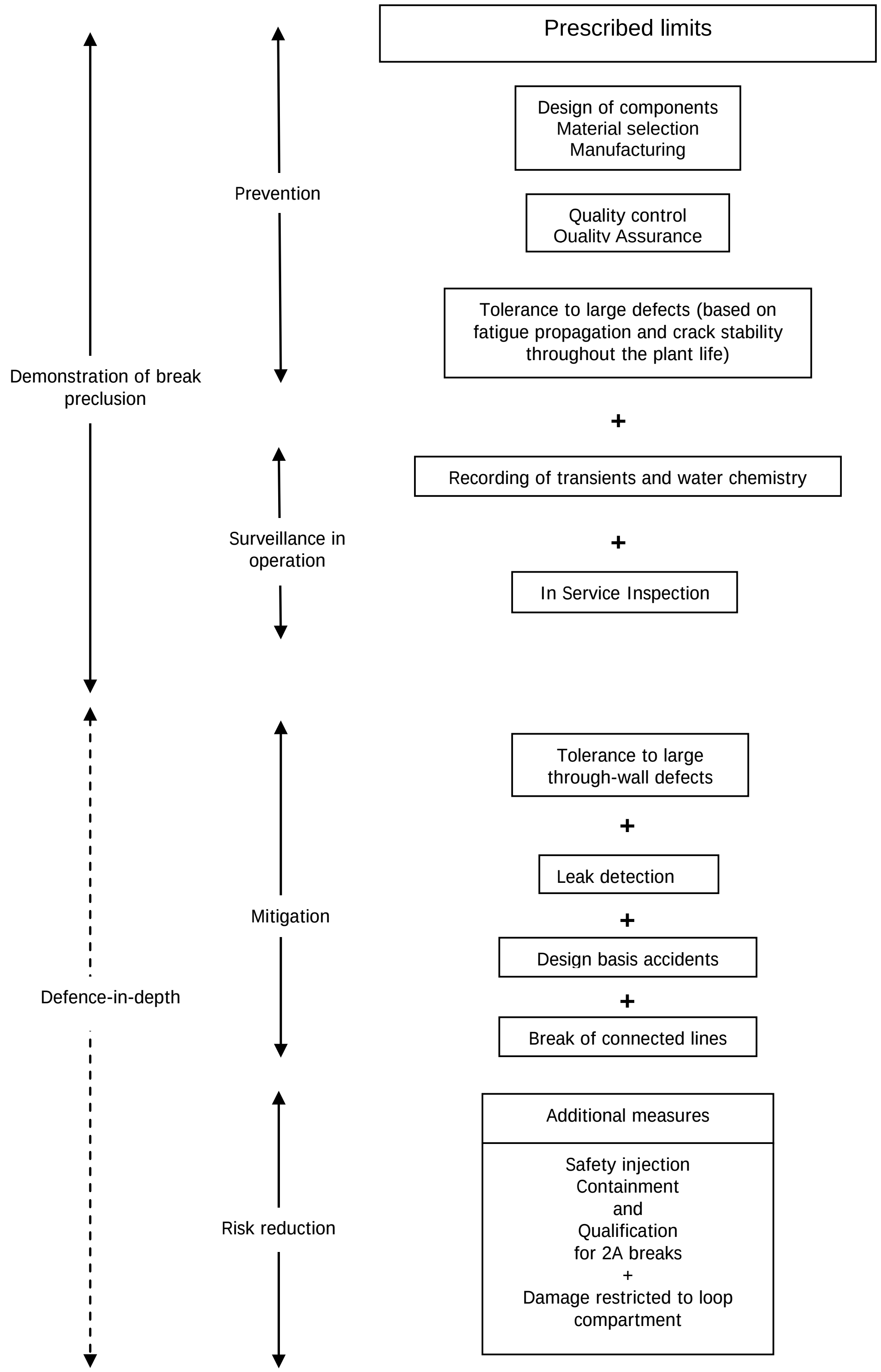
- Visual inspection
- Televisual inspection
- Acoustic surveillance

5.5. HYDROSTATIC TEST

In In-Service Inspection frame, hydrostatic tests will be carried out as required by UK regulations or requirements.

SUB-CHAPTER 5.2 - TABLE 1

Overview of the Break Preclusion principle



SUB-CHAPTER 5.2 - TABLE 2

Break Preclusion principle assumptions for postulated pipe breaks

Impact		**Postulated pipe break**
on	**of**	
Core behaviour	Loss of coolant	Break of the surge line
Fuel assemblies	Dynamic effects of decompression	Break of main coolant line connected pipework
Primary components internals (RPV, SG, Pump flywheel)	Dynamic effects of decompression	Break of main coolant line connected pipework
Primary components and supports	Dynamic effects of decompression	Break of main coolant line connected pipework
Containment internal structures	Differential pressure Temperature Flooding	Break of the surge line

SUB-CHAPTER 5.2 - TABLE 3
Prevention of potential damage

<table>
<tr><th rowspan="3">Damage</th><th rowspan="3">Main parameter</th><th rowspan="3">Areas considered</th><th rowspan="3">Potential risk</th><th colspan="5">Defence-in-depth level</th></tr>
<tr><th colspan="3">First level</th><th colspan="2">Second level</th></tr>
<tr><th>Materials Design</th><th>Manufacturing Inspections</th><th>Justification by analysis
Final confirmation</th><th>In-service inspection / Maintenance</th><th>Operational measures</th></tr>
<tr><td rowspan="8">GENERAL DAMAGE

Excessive / progressive deformation
Plastic instability</td><td rowspan="4">Conventional materials properties</td><td rowspan="2">Base materials</td><td>Initial properties inadequate
Creep</td><td>Materials selection
Specifications
Creep excluded</td><td>Confirmation of mechanical properties</td><td>Analysis
Hydrostatic test</td><td rowspan="4">Not applicable</td><td rowspan="2">Not applicable</td></tr>
<tr><td>Ageing
(softening, ductility loss)</td><td>Materials selections ductility ensured</td><td>Verified design assumptions</td><td>Materials file</td></tr>
<tr><td rowspan="2">Welds</td><td>Inadequate properties</td><td>Selection of filler metals
Properties ≥ BM
(Base Metal)</td><td>Properties verification</td><td>Not applicable</td><td rowspan="2">Not applicable</td></tr>
<tr><td>Ageing</td><td>Cf. BM</td><td>Cf. BM</td><td>Cf. BM</td></tr>
<tr><td rowspan="2">Geometry</td><td rowspan="2">All</td><td>Inadequate thickness</td><td>Minimum calculated thickness</td><td>Thickness and parts profile inspections</td><td>Analysis with minimum thicknesses
Volume inspection</td><td rowspan="2">Not applicable
(Thickness measurements possible within periodic tests)</td><td rowspan="2">Fluid cleanliness and chemistry monitoring</td></tr>
<tr><td>Loss of thickness in service (abrasion, generalised corrosion, wear and tear, erosion/cavitation)
Progressive deformation</td><td>Profile rules
Materials selection according to environment</td><td>Not applicable</td><td>Progressive deformation risk analysis</td></tr>
<tr><td rowspan="2">Loads</td><td rowspan="2">All</td><td>Risk of overpressure</td><td>System design: 2 independent discharge lines
+ relief valves</td><td>Not applicable</td><td>Justification with worst-case assumptions
Commissioning tests
Hydrostatic test</td><td>Periodic testing
Valve setpoint setting verification</td><td>Monitoring of reactor coolant pressure</td></tr>
<tr><td>Mechanical stress exceeded: ext. stress, restricted thermal expansion, hammering, direct and indirect internal hazards</td><td>Conservative assumptions
Excluding hammering (valve absence and reactor coolant pump inertia)</td><td>Not applicable</td><td>Justification with worst-case assumptions
Pipe displacement measurements</td><td>Pipe displacement measurements</td><td>Situation recording,
Handling according to the rules
Automated pipe reinforcement</td></tr>
</table>

SUB-CHAPTER 5.2 - TABLE 3

Prevention of potential damage (continued)

Damage	Main parameter	Areas considered	Potential risk	Defence-in-depth level				
				First level			Second level	
				Materials Design	**Manufacturing Inspections**	**Justification by analysis Final confirmation**	**In-service inspection / Maintenance**	**Operational measures**
LOCAL DAMAGE Pre-existing defect or initiated by fatigue or corrosion leading to leak or fast fracture risk after propagation	Design	Base materials	Initial properties inadequate	Materials selection (high toughness) Specifications NGOT process	Confirmation of mechanical properties	Mechanical analysis with end-of-life mechanical properties	Not applicable Materials ageing surveillance programme restricted to cases where ageing is significant)	Not applicable
			Thermal ageing	No cast products Ferrite level.	Chemical composition verification			
		Welds	Dilution abnormality	No NGT[1] process buttering	Risk excluded	Not applicable	Not applicable	Not applicable
	Manufacturing-related defects	Base materials	Lack of homogeneity	Wrought/forged used instead of cast	Reduction ratio > 3	Not applicable	Not applicable	Not applicable
			Surface cracking	Limitation of ferrite, C & B content	Verification of chemical comp. RCC-M unacceptable defect	PSI (pre-service inspection)	ISI (In-service Inspection)	Not applicable
		Homogenous welds	Density inclusions	Limit number of welded joints NGOT automatic process (Reduc. metal filler + special provisions)	Radiographic inspect. Length < 20mm	Significant hypothetical defects used in fast fracture risk and margin reports PSI	ISI density of 4 welds alternately with each VD (10 yearly outage inspection ISI) or VP (Outage Inspection), for vessel and SG connections	
			Blisters		Radiographic inspect. Length < 4mm Top < pass h			
			Lack of fusion		UT control 0° RCC-M unacceptable defect			
			High temperature cracking	Boron < 0.018%				
		Bimetallic connections	Density inclusions	NGT bimetallic connection + spec provisions (flat position weld)	Cf. Homogenous welds		Density and bimetallic welding external surface inspections for each VD or during VP	
			Blisters					
			Lack of fusion					
			High temperature cracking	Chemical comp improvement				

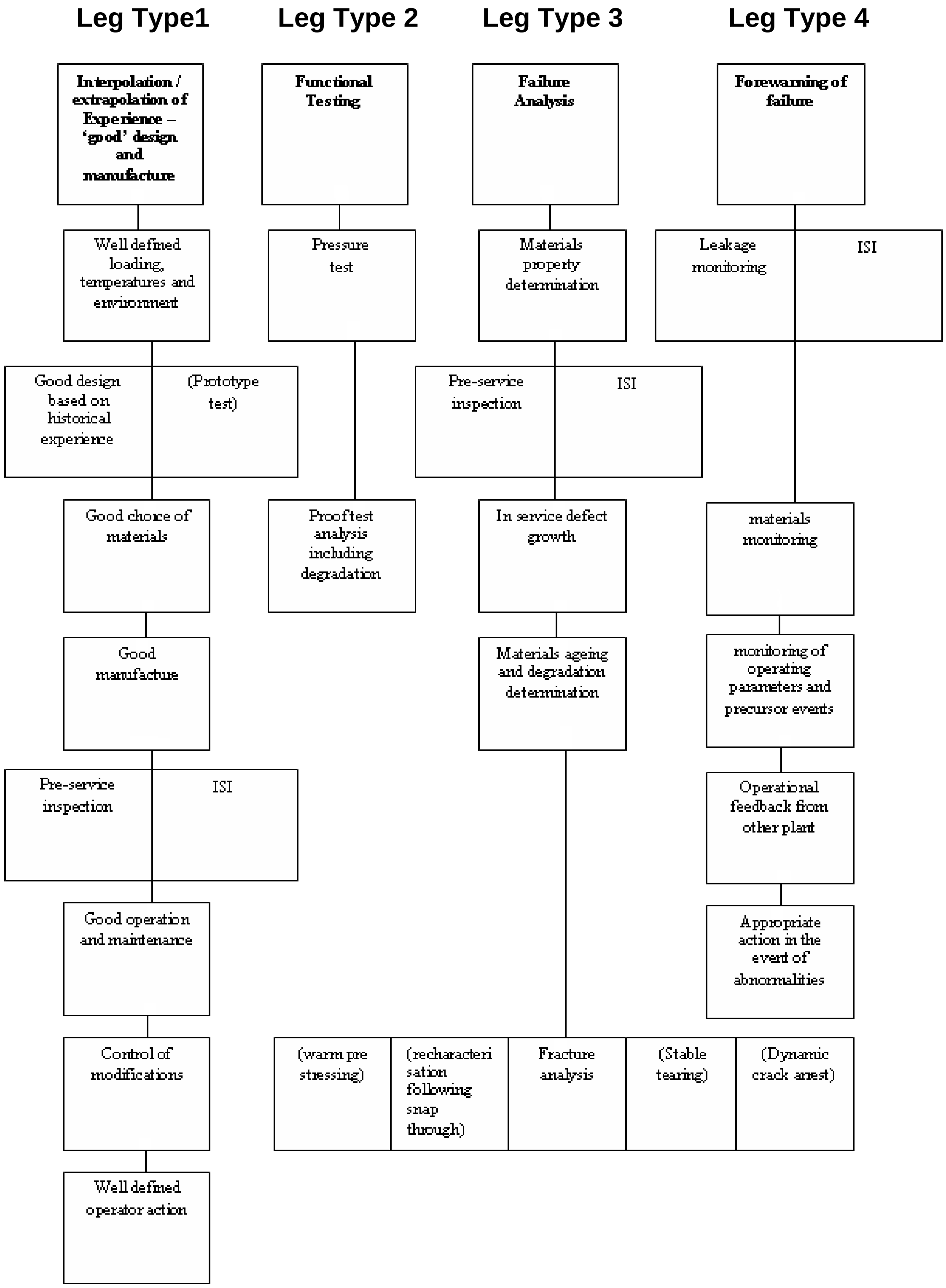
[1] Narrow Gap Tungsten inert gas welding process (NGT)

SUB-CHAPTER 5.2 –TABLE 3
Prevention of potential damage (continued)

Damage	Main parameter	Areas considered	Potential risk	Defence-in-depth level				
				First level			Second level	
				Materials Design	**Manufacturing Inspections**	**Justification by analysis Final confirmation**	**In-service inspection / Maintenance**	**Operational measures**
	Defects liable to appear in service	All	Fatigue (vibration, low cycle, thermal, stratification)	Design rules and functional requirements Vibration control	Surface condition verification	Behaviour under fatigue analysis file + integrity study Auto pressure regulation	Cf. Manufacturing-related defects	Situational recording
			Inter-granular corrosion	O2 limitation and C content	Specification verification	PSI		Chemical monitoring of the fluid
			Corrosion under stress	Limitation σ + NGT	Risk excluded	PSI		
		Bimetallic connections (LBM)	DIG (Intra-granular defaults)	NGT process	Risk excluded	PSI		Not applicable
	Loads other than fatigue and CSC	All	Design assumptions exceeded	Equipment design basis, with conservative assumptions	Not applicable	Fast fracture risk analysis under specified loads	Pipe displacement measurements	Situational monitoring Automated pipe reinforcement
			Risks of indirect damage	Physical separation and whipping drawing taken into account	Not applicable	Not applicable	Not applicable	Handling following the rules

SUB-CHAPTER 5.2 - TABLE 4

TAGSI approach to demonstrating IOF [Ref-1]



SUB-CHAPTER 5.2 - TABLE 5

Comparison of BP lines of defence-in-depth and the TAGSI legs used to demonstrate IOF

Claim for component →																	Conclusion
	Damage Prevention					**Operational Surveillance**				**Mitigation** (Prevention and limitation)				**Risk reduction**			
Break Preclusion Requirement	Design of components	Material selection	Manufacturing technique	Quality control	Tolerance to large defects (based on fatigue propagation and crack stability throughout plant life)	Pre-Service Inspection (PSI)		Surveillance of operating conditions (Recording of transients and water chemistry)	In-Service Inspection (ISI)	Tolerance to large through-wall defects	Leak detection	Tolerance to design basis accidents	Tolerance to break of connected lines	Safety injection	Containment	Qualification for 2A breaks + Damage restricted to loop compartment	**SFARP Safe component: no radioactive release, no danger to the public**
TAGSI Requirement	Good design (based on experience)	good material selection	Good manufacture		Fracture analysis	Pressure test	Proof test	Momitoring of operating parameters and precursor events	In-Service Inspection (ISI)	In-service defect growth	Leakage monitoring	Well defined loading temperatures & environment		**Not required in TAGSI**			**SFARP Safe component: no radioactive release, no danger to the public**
TAGSI leg	**1 Good design and manufacture**				**3 Failure analysis**	**2 Functional testing**		**4 Forewarning of failure**		**3 Failure analysis**	**4 Forewarning of failure**	**1 Good design and manufacture**					

SUB-CHAPTER 5.2 - TABLE 6

Comparison of requirements for 'non-breakable' components and the TAGSI legs used to demonstrate IOF

Claim for component → Conclusion

	Damage Prevention					Operational Surveillance				Mitigation (Prevention and limitation)					Risk reduction	
Non-breakable Requirement	Design of components (using experience of other facilities)	Material selection	Manufacturing technique	Quality control	Stress/failure analysis at the design stage	Non-destructive testing		Surveillance of operating conditions	In-Service Inspection (ISI)	Potential in-service degradation mechanisms considered	Tolerance to large through-wall defects	Review of experience from other facilities	Leak detection	Tolerance to design basis accidents	Risk due to gross failure shown to be acceptable in PSA	**SFARP Safe component: no radioactive release, no danger to the public**
TAGSI Requirement	Good design (based on experience)	good material selection	Good manufacture		Fracture analysis	Pressure test	Proof test	Monitoring of operating parameters and precursor events	In-Service Inspection (ISI)	Materials ageing and degradation determination	In-service defect growth	Operational feedback from other plant	Leakage monitoring	Well defined loading temperatures & environment	Not required in TAGSI	**SFARP Safe component: no radioactive release, no danger to the public**
TAGSI leg	**1 Good design and manufacture**				**3 Failure analysis**	**2 Functional testing**		**4 Forewarning of failure**		**3 Failure analysis**		**4 Forewarning of failure**		**1 Good design and manufacture**		

SUB-CHAPTER 5.2 - TABLE 7

Assumptions additional to the break preclusion procedure on the main reactor coolant system pipework

Effects		Postulated pipework failures
On	**from**	
RIS [SIS] Performance	Loss of coolant	Leak/break on the main primary pipework up to 2A break
Containment	Pressure Temperature	2A break on the main reactor coolant system pipework
Environmental qualification of equipment	Flooding Pressure Temperature Humidity Radiation	2A break on the main reactor coolant system pipework
Primary components (with internal equipment and supports)	Dynamic effects of decompression	Guillotine break of all lines connected at the connection weld
Internal containment structures	Differential pressure Temperature Flooding	2A break on the main primary pipe Guillotine break of all lines connected at the connection weld
Main component supports	Co-linear 2pA force with the nozzle	“2PA” Force

SUB-CHAPTER 5.2 – REFERENCES

External references are identified within this sub-chapter by the text [Ref-1], [Ref-2], etc at the appropriate point within the sub-chapter. These references are listed here under the heading of the section or sub-section in which they are quoted.

3. HIGH INTEGRITY COMPONENT REQUIREMENTS

3.1. BREAK PRECLUSION PIPING REQUIREMENTS

3.1.1. Main requirements for the demonstration of Break Preclusion

[Ref-1] Application of the break preclusion assumption in the main reactor coolant and steam lines of the EPR FA3. ECEMA040920 Revision C1. EDF. (E)

[Ref-2] Break Preclusion in reactor main coolant lines and main steam lines. Positioning of the concept and associated safety requirements. ENSNDR080245 Revision A. EDF. (E)

3.1.1.1. Preventive measures

3.1.1.1.2. Prevention of local damage

[Ref-1] EPR RF 002 - Alloy 690 material data file. NFPMT DC 39 Revision C. AREVA. December 2004. (E)

3.1.2. Requirements for further lines of defence-in-depth

3.1.2.1. First further line of defence-in-depth

3.1.2.1.2. Leak detection via the main reactor coolant system

[Ref-1] Break Preclusion in reactor main coolant lines and main steam lines. Positioning of the concept and associated safety requirements. ENSNDR080245 Revision A. EDF. (E)

3.1.2.2. Second further line of defence-in-depth

3.1.2.2.2. Design basis accidents

[Ref-1] Break Preclusion in reactor main coolant lines and main steam lines. Positioning of the concept and associated safety requirements. ENSNDR080245 Revision A. EDF. (E)

3.3. COMPARISON OF REQUIREMENTS FOR BREAK PRECLUSION / NON BREAKABLE COMPONENTS WITH UK REQUIREMENTS FOR IOF COMPONENTS

3.3.2. Requirements for UK safety cases for IOF

[Ref-1] UK Health and Safety Executive (HSE). Safety Assessment Principles for Nuclear Facilities. 2006 Edition Revision 1. January 2008 (E)

[Ref-2] UK Health and Safety Executive (HSE). Technical Assessment Guide. Integrity of Metal Components and Structures. TAG T/AST/16 Issue 3. August 2008 (E)

[Ref-3] 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, 2001, 539 – 552 (E)

3.3.3. Comparison of the Break Preclusion principle with the TAGSI approach for IOF

[Ref-1] Break Preclusion in reactor main coolant lines and main steam lines. Positioning of the concept and associated safety requirements. ENSNDR080245 Revision A. EDF. (E)

3.3.4. Comparison of requirements for non-breakable components with the TAGSI approach for IOF

[Ref-1] Design and Construction Rules for mechanical components of PWR nuclear islands (RCC-M). (See RCC-M Subsection B). AFCEN. 2007 edition supplemented by RCC-M Modification Sheet FM 1060. (E)

5. IN-SERVICE INSPECTION OF THE REACTOR COOLANT SYSTEM

5.3. ACCESS NECESSARY TO INSPECT THE RCPB

[Ref-1] F Chavigny. Demonstration of the accessibility and controllability for in-service inspection of the structural integrity components. ECEMA101028 Revision A. EDF. April 2010. (E)

[Ref-2] Ultrasonic examination of MCL homogeneous and dissimilar metal welds. PEEM-F 11.0505 Revision C. AREVA. March 2012. (E)

SUB-CHAPTER 5.2 - TABLE 4

[Ref-1] 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, 2001, 539 – 552 (E)