Office for Nuclear Regulation

ONR GUIDE			
INTEGRITY OF METAL STRUCTURES, SYSTEMS AND COMPONENTS			
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1. INTRODUCTION

1.1 ONR has established its Safety Assessment Principles (SAPs). The principles presented in the SAPs are supported by a suite of technical assessment guides (TAGs) to assist ONR inspectors in their assessments to support regulatory judgments and decisions. This document is one of these guides.

1.2 The TAG has been revised to reflect discussions with licensees and requesting parties during a workshop organised by ONR and the Forum for Engineering Structural Integrity (FESI) to develop a consistent understanding of the expectations for highest reliability nuclear pressure systems [1]. It was noted at the workshop that ONR guidance had been developed for the assessment of submissions from a single mature licensee and that, with new nuclear build, the number and variety of stakeholders — including licensees, prospective licensees and requesting parties — had increased. Therefore, the implicit assumptions that had been developed for highest reliability plant warranted further explanation, particularly on the adequacy of margins in defect tolerance assessments (DTAs) and the expectations for, and benefits of, inspection qualification (IQ). In response, this revision of the TAG provides further guidance on:

- Expectations for highest reliability plant, in particular, on the adequacy of margins in DTAs consistent with the Sizewell B inquiry and Generic Design Assessment (GDA) findings and regulatory observations (§5.91 to §5.96 & Appendix A4)
- Expectations for, and benefits of, inspection qualification (Appendix A5).

This revision of the TAG updates existing guidance and includes additional guidance on the use of the Master curve approach to predict fracture toughness (§5.99 & §5.101), welding (§5.66 & 5.67) and ageing and degradation (§5.116 to 5.125).

2. PURPOSE AND SCOPE

2.1 This TAG provides ONR inspectors with guidance and interpretation of the SAPs concerned with the integrity of metallic structures, systems and components (SSCs) that supplements guidance in the SAPs [2], in particular, EMC.1 to EMC.16 & EMC.18 to EMC.34 and those on ageing and degradation, EAD.1 to EAD.5.

2.2 The scope of the TAG excludes metallic structures within the ambit of civil engineering such as building frames, pipe bridges and crane supports.

3. RELATIONSHIP TO LICENCE AND OTHER RELEVANT LEGISLATION

3.1 The primary licence conditions (LCs) for which assessments of metal SSCs are to be carried out are:

- LC 14 (Safety documentation)
- LC 15 (Periodic review)
- LC 17 (Management systems)
- LC 19 (Construction or installation of new plant)
- LC 20 (Modification to design of plant under construction)
- LC 21 (Commissioning)
- LC 22 (Modification or experiment on existing plant)
- LC 23 (Operating rules)
- LC 24 (Operating instructions)
- LC 25 (Operational records)
- LC 26 (Control and supervision of operations)
- LC 27 (Safety mechanisms, devices and circuits)
- LC 28 (Examination, inspection, maintenance and testing)

- LC 29 (Duty to carry out tests, inspections and examinations)
- LC 30 (Periodic shutdown)
- LC 34 (Leakage and escape of radioactive material and radioactive waste)
- LC 35 (Decommissioning)

3.2 Other licence conditions relevant to metal SSCs are:

- LC 6 (Documents, records, authorities and certificates)
- LC 10 (Training)
- LC 12 (Duly authorised and other suitably qualified and experienced persons)
- LC 13 (Nuclear safety committee)
- LC 36 (Organisational capability)

3.3 Inspectors should note there is other relevant legislation such as Pressure Systems Safety Regulations (PSSR) 2000; see NS-TAST-GD-067 (Pressure Systems Safety).

4. RELATIONSHIP TO SAPS, TAGS, WENRA REFERENCE LEVELS AND IAEA SAFETY STANDARDS

SAPs

The ONR SAPs for the integrity of metal SSCs are EMC.1 to EMC.34 (paragraphs 280 to 319) which are closely related to those for ageing and degradation: EAD.1 to EAD.5 (paragraphs 212 to 221). These SAPs—in the sequence adopted by [2]—are:

<u>Integrity of metal SSCs</u>

- Highest reliability structures or components
  - EMC.1 (Safety case and assessment)
  - EMC.2 (Use of scientific and technical issues)
  - EMC.3 (Evidence)
- General
  - EMC.4 (Procedural control)
  - EMC.5 (Freedom from and tolerance of defects)
  - EMC.6 (Means to identify defects)
- Design
  - EMC.7 (Loadings)
  - EMC.8 (Providing for examination)
  - EMC.9 (Product form)
  - EMC.10 (Weld positions)
  - EMC.11 (Failure modes)
  - EMC.12 (Brittle behaviour)
- Manufacture and installation
  - EMC.13 (Materials)
  - EMC.14 (Techniques and procedures)
  - EMC.15 (Control of materials)
  - EMC.16 (Contamination)
  - EMC.18 (Third-party inspection)
  - EMC.19 (Non-conformances)
  - EMC.20 (records)
- Manufacturing, pre-service and in-service examination and testing
  - EMC.27 (Examination)
  - EMC.28 (Margins)
  - EMC.29 (Redundancy and diversity)
  - EMC.30 (Qualification)

- Operation
  - EMC.21 (Safe operating envelope)
  - EMC.22 (Material compatibility)
  - EMC.23 (Ductile behaviour)
- Monitoring
  - EMC.24 (Operation)
  - EMC.25 (Leakage)
  - EMC.26 (Forewarning of failure)
- In-service repairs and modifications
  - EMC.31 (Repairs and modifications)
- Analysis
  - EMC.32 (Stress analysis)
  - EMC.33 (Use of data)
  - EMC.34 (Defect sizes)

<u>Ageing and degradation</u>

- EAD.1 (Safe working life)
- EAD.2 (Lifetime margins)
- EAD.3 (Periodic measurement of material properties)
- EAD.4 (Periodic measurement of parameters)
- EAD.5 (Obsolescence).

The sequence of the SAPs on the integrity of metal SSCs in [2] (and §4.1) reflects changes to the SAPs and editorial policy.

- Advice on examination during manufacture which was given in EMC.17 in the previous edition of the SAPs is now given in EMC.27 to EMC.30.
- Consequently, EMC.17 has been removed.
- SAPs EMC.7 to EMC.16 and EMC.18 to EMC.34 are given in the sequence of processes applied to a component.
- Editorial policy for the SAPs is to retain numbers for SAPs that are largely unchanged.

Where assessment is for structural integrity of an SSC which forms part of a containment, the SAPs on containment and ventilation, ECV.1 to ECV.10 should be considered. Where the SSC forms part of a core support structure, SAPs ERC.1 to ERC.4 should be considered.

Other relevant SAPs include:

- ECS.1 to ECS.5 (Safety classification and standards)
- EDR.1 to EDR.3 (Design for reliability)
- EHA.1 to EHA.19 (External and internal hazards)
- EKP.1 to EKP.5 (Key principles)
- ELO.1 (Layout - access)
- EMT.1 to EMT.8 (Maintenance, inspection and testing)
- EPS.1 to EPS.5 (Pressure systems)
- EQU.1 (Equipment qualification)
- ERL.1 & ERL.2 (Form of claims) and paragraphs 190 to 193
- FA.2, FA.5 to FA.9 (Fault analysis) and paragraph 656, in particular 656(c)
- NT.1 (Numerical targets)
- SC.1 to SC.8 (Safety case processes)

Relevant TAGs

4.1 TAGs that are relevant to structural integrity assessments include:

- NS-TAST-GD-005 Guidance on the demonstration of ALARP (As Low As Reasonably Practicable)
- NS-TAST-GD-009 Examination, inspection, maintenance and testing of items important to safety
- NS-TAST-GD-013 External hazards
- NS-TAST-GD-014 Internal hazards
- NS-TAST-GD-026 Decommissioning of nuclear licensed sites
- NS-TAST-GD-030 Probabilistic safety analysis
- NS-TAST-GD-033 Duty-holder management of records
- NS-TAST-GD-042 Validation of computer codes and calculation methods
- NS-TAST-GD-049 Licensee core safety and intelligent customer capabilities
- NS-TAST-GD-051 The purpose, scope and content of nuclear safety cases
- NS-TAST-GD-050 Periodic Safety Reviews (PSR)
- NS-TAST-GD-067 Pressure Systems Safety
- NS-TAST-GD-077 Supply chain management arrangements for the procurement of nuclear safety related items or services
- NS-TAST-GD-088 Chemistry of operating civil nuclear reactors
- NS-TAST-GD-089 Chemistry assessment
- NS-TAST-GD-094 Categorisation of safety functions and classification of structures systems and components
- NS-TAST-GD-098 Asset management

WENRA reference levels and IAEA safety standards

4.2 Relevant Western European Nuclear Regulators Association (WENRA) Reactor Safety, Decommissioning and Waste and Spent Fuel Storage Reference Levels and International Atomic Energy Agency (IAEA) standards, guidance and technical documents are considered in Appendices A1 & A2 respectively.

5. ADVICE TO INSPECTORS

Introduction

5.1 This document provides advice on the assessment of the structural integrity aspects of safety cases for metallic SSCs. The relevant SAPs (§4.1 to §4.4) provide a framework for the inspector to judge the adequacy of the structural integrity aspects of a safety case. The relevance and importance of the various SAPs will vary between safety cases. For example, SAPs EMC.1, EMC.2 and EMC.3 give advice on safety cases for highest reliability SSCs where gross failures can be discounted.

5.2 A safety case for metal SSCs should be examined in the context of the overall safety case for the plant taking account of interactions with other safety features. There may be defence in depth that can protect or mitigate the effect of failure to a greater or lesser degree. It may be that the direct effect of structural failure is trivial but the indirect consequences may be failure of safety related plant, instrumentation, or operator dose uptake, i.e. the failed part acts as an internal hazard to other safety features. Moreover, safety cases for the structural integrity of nuclear sites are often complex and can require assessment by a range of specialists to judge whether they are adequate and ALARP. Therefore assessment of the structural integrity aspects of a safety case is rarely undertaken in isolation.

5.3 The starting point for a structural integrity assessment is the categorization of functions (ECS.1) and the safety classification of an SSC (ECS.2).

Categorisation of safety functions and safety classification of SSCs

5.4 Guidance on safety classification, codes and standards is provided in ECS.1 to ECS.5, paragraphs 158 to 173 of the SAPs. The purpose of an ONR assessment of a structural integrity safety case is to come to a view on whether it is adequate and ALARP for all the conditions stated in the case noting that those conditions need to cover the design basis including fault conditions. The inspector needs to begin by understanding the safety functions delivered by the facility (ECS.1); this will determine the safety classification of the SSC (ECS.2), the appropriate codes and standards (ECS.3) and any other additional measures. On ECS.1, SAPs paragraph 160 states:

> *'The safety categorisation scheme employed should be linked explicitly with the licensee's design basis analysis (see paragraph 607). Various schemes are in use in the UK; these principles have been written assuming categorisation on the following basis:*
>
> *(a) Category A — any **function** that plays a principal role in ensuring nuclear safety*
>
> *(b) Category B — any **function** that makes a significant contribution to nuclear safety.*
>
> *(c) Category C — any other safety **function** contributing to nuclear safety.'*

On safety classification ECS.2 paragraph 166 states:

> *'A number of different safety classification schemes are in use in the UK. The following scheme, linked to the categorisation scheme outlined in paragraph 160, is recommended in these principles:*
>
> *(a) Class 1 – any **structure, system or component** that forms a principal means of fulfilling a Category A safety function.*
>
> *(b) Class 2 – any **structure, system or component** that makes a significant contribution to fulfilling a Category A safety function, or forms a principal means of ensuring a Category B safety function.*
>
> *(c) Class 3 – any **other structure, system or component** contributing to a categorised safety function.'*

Further guidance on nuclear safety classification is given in paragraphs 158 to 177 of the SAPs and TAG NS-TAST-GD-094 (Categorisation of safety functions and classification of structures systems and components).

5.5 The categorisation of the safety function and the safety classification of the SSC determine the requirements for design, manufacture, construction, installation, operation, monitoring, inspection, maintenance and testing. For example, the catastrophic failure of a reactor pressure vessel (RPV) of a large power plant would almost certainly lead to unacceptable radiological consequences; hence, the highest standards are required at each stage of the life of such a vessel. A claim that primary cooling circuit pipework will not suffer guillotine type failures might also fall into this category. On the other hand, the radiological consequences of initial leakage from certain chemical plant containment may be less significant, provided that there is confidence in double containment to allow detection of and recovery from the situation. In the latter case, appropriate industrial, national or international standards may be sufficient. Thus, an appreciation of the consequences of failure forms the basis of an assessment of structural integrity. SAPs paragraphs 286 to 296 are the basis for assessment of the situations demanding the highest integrity and paragraphs 297 to 300 summarize the approach for less demanding situations.

5.6 The safety functional requirements of SSCs should be identified from the fault schedule; see SAPs paragraphs 407 and 643, and the appropriate safety classification determined in accordance with principles ECS.1, ECS.2 and associated paragraphs. In general, the safety functional requirement of an SSC will depend on the potential radiological consequences of its failure (ECS.1), and the requirement to meet the functional requirement for the proposed life of the facility (ECS.2). ECS.2 requires that SSCs should be categorised based on the consequences of failure and of the failure frequency requirements of the safety case. From this, the appropriate standards of design, manufacture, installation and testing, in-service maintenance, inspection and testing, and operation can be derived (ECS.3). The inspector should therefore verify the potential radiological consequences of structural failure at an early stage in the assessment process to enable the depth and breadth of the assessment to be established. It is also important that the licensee has identified failure modes and likelihoods. The failure modes should be ranked in terms of the significance of the consequences. In addition, there may be a need to consider industrial safety; this may be covered by the nuclear safety requirements, or need explicit consideration.

5.7 IAEA Safety Standards Series Requirements document SSR 2/1 (Appendix A2) [A2 2] defines three fundamental safety functions:

> *'Fulfilment of the following fundamental safety functions for a nuclear power plant shall be ensured for all plant states: (i) control of reactivity, (ii) removal of heat from the reactor and from the fuel store and (iii) confinement of radioactive material, shielding against radiation and control of planned radioactive releases, as well as limitation of accidental radioactive releases.'*

The second and third fundamental safety functions set the requirements for a through life structural integrity safety case. Several of the safety requirements in [A2.2] are relevant for the integrity of metal SSCs including inter alia:

- Requirement 9: Proven engineering practices
- Requirement 10: Safety assessment
- Requirement 15: Design limits
- Requirement 22: Safety classification
- Requirement 28: Operational limits and conditions for safe operation
- Requirement 31: Ageing management
- Requirement 44: Structural capability of the reactor core boundary.

Numerical targets, BSLs & BSOs

5.8 SAP NT.1 Targets 8 & 9 give upper limits on risk of exposure to ionising radiation (the Basic Safety Levels [BSLs]) and upper limits on the broadly acceptable region (the Basic Safety Objectives [BSOs]). Often it is not possible to apply such quantification and instead assessment will be more in terms of qualitative likelihood, SAPs paragraphs 291 and 656(c). Nevertheless, the concept of a band between an indicative upper limit on risk and broadly acceptable risk is useful; it is the region in which as low as reasonably practicable (ALARP) is given priority by inspectors (noting that legally there is a need to reduce all risks SFAIRP). Inability to meet a BSL should only be an issue for existing plant. It is ONR policy that a new installation should at least meet BSLs; see SAPs paragraph 698.

5.9 If it seems that a BSL is exceeded, the inspector should carefully read SAPs paragraphs 698 and 699. If a BSL is clearly exceeded and there is no prospect of improvement in the long-term, the issue moves into consideration under the relevant Enforcement Management Model (EMM) [3] which is outside the scope of this guidance; the inspector should consult ONR guidance on probabilistic safety analysis (TAG NS-TAST-GD-030). The inspector may also need to consider some of the ONR compliance procedures in terms of the potential outcome of the assessment including the Enforcement Policy Statement [4]. If the inspector judges that a BSL is comfortably exceeded and the EMM indicates that significant regulatory action should be taken (e.g. shutdown), ONR management will need to be engaged and convinced of any proposed action.

Structural integrity safety cases

5.10 The starting point for design is compliance with relevant national and international codes and standards. In addition, depending on the nuclear safety significance, safety case claims for the structural integrity of SSCs may require further substantiation.

5.11 The general lack of adequate reliability data, particularly for higher reliability SSCs, leads to assessment being based primarily on established deterministic engineering practice. Even when there is some confidence in assessing reliability based on existing data and a probabilistic safety case is possible, it is unlikely to be acceptable without substantial evidence-based support. As a result, although the radiological consequences of failure of structural components may be significant, the use of a Probabilistic Safety Analysis (PSA) might be indicative or nominal compared with other aspects of the PSA; see NS-TAST-GD-030 for more information on PSA.

5.12 For safety cases supported by more than one argument (multi-legged safety cases), each argument (leg) needs to be considered separately before coming to a view on the overall adequacy of the safety case. Due consideration should be given to the potential for common mode failure mechanisms and factors that affect more than one argument (see NS-TAST-GD-051 for advice on safety case assessments).

Highest reliability SSCs

5.13 Appendix A4 outlines the development of the UK expectations for highest reliability SSCs following the Sizewell B inquiry and the experience from the Generic design assessment (GDA) process.

5.14 In some cases, a licensee may propose a safety case where the likelihood of gross structural failure is claimed to be so low that it may be discounted; even so, if failure did occur, the consequences would be unacceptable. Licensees invoke such lines of argument where the consequences are unacceptable or where it would be difficult to demonstrate that consequences are acceptable. One reason for unacceptable consequences is often that there is no means of protecting or mitigating the effects of the failure. In the UK, this is often referred to as there being no 'line of protection'. ONR does not seek or encourage this basis for a safety case in any particular circumstance; even so, ONR will assess such cases on their merits.

5.15 To assess a safety case for highest reliability plant, the inspector should consider the relevant SAPs (paragraphs 280 to 300) to the appropriate depth to establish whether the evidence to support the claims (as listed in §5.24) provides the necessary confidence that the safety functional requirements will be met.

5.16 In terms of limits on risk for cases that discount gross failure, the SAPs Target 9 (Total risk of 100 or more fatalities) is the most relevant. However, the ONR inspector should be aware that the potential consequences of a gross structural integrity failure could exceed the Target 9 levels. Depending on inventory and accident sequence, release quantities of order up to 100 times those implied in Target 9 could be relevant. Given the linear relationship between consequence (dose, release) and frequency in SAPs Targets in general, consequences 100 times greater than those in Target 9 imply a requirement for frequency of occurrence to be 100 times lower. There is also the question of whether a single class of accident should contribute more than a fraction of total risk (see footnote to SAPs Target 8 and consider it for Target 9). The ONR inspector may find it useful to approach assessment of a safety case that discounts gross failure from this perspective. For existing plant, a less attractive possibility would be to assess the structural integrity safety case by comparison with the overall perceived risk from the installation. However, this approach is the least favoured. Moreover, cliff-edge effects should be considered; a small change in design basis fault or event assumptions should not lead to a disproportionate increase in radiological consequences (EHA.7).

5.17 The LC14 licence condition for a UK licensee to have an adequate safety case includes highest reliability SSC expectations that extend beyond those of the design codes. These expectations, which are more open to interpretation than code compliance, include third-party surveillance, inspection qualification, repeat inspection, compositional checks, mechanical testing and fracture mechanics assessments. There are challenges in meeting these expectations.

5.18 A case that claims gross failure is so remote it may be discounted carries a high burden of proof (arguments and evidence). Such a case cannot be made by simple assertion of the robustness of an SSC. So declaring an SSC to have this status is not to be seen as an easy option simply to avoid considering the consequences of failure, i.e. as a time-saver in the hazard/consequences area. Discounting gross failure should only be invoked if the consequences of failure are unacceptable or it is not possible to demonstrate that the consequences are acceptable. SAPs paragraphs 287 to 291 discuss such safety cases. The content of the SAPs will not be repeated here. However, the following are emphasised:

> *'…a claim that gross failure of a pressure vessel may be discounted cannot be plausibly associated with a failure rate much better than 1×10^{-7} to 1×10^{-8} per vessel year…'*
>
> *'…claims for pipework weld failure rates for gross failure (e.g. guillotine failure) much better than 1×10^{-8} to 1×10^{-9} per weld year should not be considered plausible…'*

The SAPs do not give indicative failure frequencies for highest reliability (or other components). It would be for a licensee to define these in its safety case based on established norms. These statements indicate failure frequencies beyond which ONR would not consider the claims to be credible. A safety case that discounts gross failure cannot be a 'formal proof' of such reliability levels. See §5.133 to §5.136 for guidance on reliability statements based on operational experience.

5.19 The aim in assessing a structural integrity safety case that discounts gross failure is not to check for 'perfection' in every individual aspect. Rather the main aim of the assessment of a safety case that discounts gross failure is to check that a claim of very high reliability/quality is met for all aspects and that there is sufficient defence-in-depth in the array of structural integrity measures and arguments. The aim is that an individual aspect which is short of 'perfection' cannot by itself precipitate gross failure. It is the extent of structural integrity reliability/quality and defence-in-depth in the safety case evidence that distinguishes a case that discounts gross failure from structural integrity safety cases that claim to substantiate a lower level of reliability.

5.20 A safety case that discounts gross failure will attract commensurate ONR assessment interest. Usually, structural integrity cases that discount gross failure will imply a level of reliability higher than is demonstrable by actuarial statistics (see SAPs paragraphs 291 & 656(c)). In judging a case that discounts gross failure, the inspector should bear in mind ONR policy that a new facility should at least meet the BSLs and there is a level of broadly acceptable risk. The limit on tolerability of risk in this case is effectively the minimum set of conditions to apply which make a claim of discounting gross failure plausible. ALARP is relevant to ways of improving the case beyond this minimum set of conditions; ALARP is not relevant to arguing acceptance of a case that does not meet the judged minimum set of conditions.

5.21 As background to the assessment of structural integrity safety cases that discount gross failure, the inspector may wish to consider the Technical Advisory Group Structural Integrity (TAGSI) response to ONR questions [5].

5.22 The highest demands are placed on a structural integrity safety case when the consequences of failure would be extreme and the licensee claims that the likelihood of gross failure is so low that it may be discounted (SAPs paragraphs 286 to 296). This type of safety case is discussed in the context of Sizewell B in [6] and [7].

5.23 EMC.1 addresses two particularly important aspects, that the SSC:

- Should be as defect free as possible and
- Be tolerant of defects.

This wording which is in terms of crack-like defects can be generalized; the SAPs use the term 'defect' for any significant deviation from nominal, covering inter alia, crack-like defects, wall thinning, creep damage and dimensional deviations. In principle, a component could fail due to overload without any contribution from degradation in the fabric of the component. The SAPs deprecate the type of structural integrity safety case that discounts gross failure; see paragraph 286; however, the SAPs accommodate the likely necessity to assess these safety cases in some circumstances.

5.24 Suggested evidence to demonstrate highest reliability requirements is given in SAPs EMC.3, paragraph 295:

(a) the use of sound design concepts and proven design features;
(b) a detailed design loading specification covering normal operation, faults and accident conditions. This should include plant transients and internal and external hazards;
(c) consideration of potential in-service degradation mechanisms;
(d) analysis of the potential failure modes for all conditions arising from design specification loadings;
(e) use of proven materials;
(f) confirmatory testing to demonstrate that the parent materials and welds have the appropriate material properties, especially strength and the necessary resistance to fracture;
(g) application of high standards of manufacture, including manufacturing inspection and examination;
(h) high standards of quality management throughout all stages of design, procurement, manufacture, installation and operation (see also paragraph 207 of the SAPs on excluding foreign material);
(i) pre-service and in-service examination to detect and characterise defects at a stage before they could develop to cause gross failure;
(j) defined limits of operation (operating rules), supported as necessary by safety measures (e.g. overpressure protection);
(k) in-service monitoring of facility operational parameters;
(l) in-service materials monitoring schemes;
(m) a process for review of facility operation to ensure the facility is operated and materials performance is within the assumptions of the safety case;
(n) a process for review of and response to deviations;
(o) a process for review of experience from other facilities, developments in design and analysis methodologies and the understanding of degradation mechanisms for applicability to the component or structure in question; and
(p) a process for control of in-service repairs or modifications to similar codes, specifications and standards as for original manufacture, taking account of developments since manufacture.

5.25 Much of this evidence contributes to avoiding defects and the defect tolerance of SSCs, or the management of aspects that affect defect tolerance. For an overview of an example of the elements of a safety case that discounts gross failure and that has been assessed by ONR, see [6] and [8]. For guidance on DTAs, see §5.91 through §5.96.

5.26 Where gross failure is discounted by invoking a highest reliability claim, ONR expects a demonstration to show that highest reliability SSCs are not unduly challenged by the consequences of postulated gross failure of other SSCs e.g. the internal hazards arising from pipe-whip and missiles (NS-TAST-GD-014 – internal hazards).

SSCs of other than the highest reliability

5.27 The integrity of SSCs for which a safety case does not discount the possibility of gross failure carries a lesser burden of requirements than those for which the highest reliability claims are made. For lower safety class SSCs, compliance with appropriate national and international standards may be sufficient.

5.28 In this case, the list of requirements in SAPs paragraph 295 (EMC.3), i.e. items (a)-(p) above, is also relevant though the stringency of their application should reflect the lower safety classification of the SSC (see SAPs paragraphs 297 to 300). SAPs EMC.4 to EMC.34 are however applicable to these SSCs.

Design - The use of sound design concepts and proven design features

5.29 To demonstrate that structures meet their safety functional requirements it is necessary to establish that sound design concepts, rules, standards, methodologies and proven design features have been used, and that the design is robust. The design requirements depend on the safety classification of the SSC. For guidance on safety categorisation, classification and codes and standards, the inspector should refer to paragraphs §5.4 to §5.7 of this TAG. Design activities should be subject to procedural control (EMC.4).

5.30 The design of some SSCs might not be based on any recognized published design code. In this case, the inspector should examine the justification provided by the licensee to establish that it is based on sound scientific understanding, and that the design methods are supported by suitable experimental verification and validation. As required by ERL.1 and paragraphs 191(a), (b) and (d), the safety case should include a comprehensive examination of all the relevant scientific and technical issues.

5.31 Designs should be supported by appropriate research and development and any novel features adequately tested before coming into service, and subsequently monitored during service, SAPs paragraph 281.The data used in analyses and acceptance criteria should be clearly conservative, taking account of uncertainties in the data and their contribution to the safety case (EMC.33).

Design codes

5.32 In general, the expectation is that — whenever practicable — the most recent version of a design code is used. An assessment of plant where this is not realistic may necessitate a gap analysis against the current version of a code.

5.33 Any deviation from the code should be justified since design codes are developed with implicit safety factors and assume minimum material properties and quality of fabrication. Design codes provide a holistic method of assessment of steel components. For example, a less stringent thickness requirement might be compensated by the flexibility of a joining component. The inspector should ensure that all areas of the code have been complied with and that the whole process has been followed. Although compliance with a design code provides a high level of confidence in the structural integrity of steel components against the design conditions, for highest reliability components, additional considerations and/or analysis will often be required; see requirements for highest reliability SSCs in §5.13 to §5.26.

5.34 For pressure boundary and other load bearing structures, the use of historic standards might be acceptable as a minimum. However, where codes are perceived not to reflect modern requirements or practices, it may be worthwhile and practicable to invoke additional stress analysis and analysis of fabrication processes, inspections or materials.

5.35 The design concept should incorporate appropriate protection systems and monitoring systems to enable the SSC to be maintained within its safe operating envelope for the duration of the life of the installation. For pressure boundary SSCs, these would typically include overpressure protection systems, thermocouples for monitoring temperatures, safety relief valves, leak detection systems, loss of coolant feed trip systems. For other load bearing structures, the emphasis would probably be more on monitoring systems. Adequate arrangements need to be in place for maintenance, inspection, and testing of the monitoring systems to ensure that the safety functional requirements continue to be met.

Verification requirements

5.36 The licensee's process for developing the safety case should include adequate checking, verification and independent review to a degree appropriate to the case (MS.2, MS.4, EHF.8 and paragraphs 58, 60-68, 77, 98 & 457 of the SAPs). The starting point for design is compliance with relevant national and international codes and standards. In addition, depending on the nuclear safety significance, safety case claims for the structural integrity of SSCs may require further substantiation.

5.37 This may be particularly demanding for existing metal SSCs where, by comparison with modern standards, shortcomings may be present in some aspects of the argument and it may not be possible to introduce changes; see paragraph 31 of the SAPs [2].

5.38 Other measures, such as changes to operating conditions, may be necessary to achieve an acceptable safety case. In some cases, consideration should be given to the reasonable practicability of enhancing confidence in the safety case by additional research, examination, measurements, material examination, analysis, or enhanced monitoring or by making alternative provisions to ensure safety.

Changes in design codes and standards

5.39 For existing plant, it is recognised that the original design codes and standards may have changed, and other factors such as additional loads, degradation mechanisms, or advances in analysis methods may enhance or erode some of the explicit and implicit safety margins in codes. It may be necessary to check for significant changes in codes through time (e.g. manufacturing examinations before or after post weld heat treatment). It should be established that the original design codes and standards remain appropriate and that their application is consistent. Any deviation from the code should be demonstrated to be acceptable for the overall safety case. This aspect can give rise to difficulties for pressure vessels and pipework systems, particularly in the case of fault loads or unforeseen degradation mechanisms that were not addressed at the design stage. The PSR provides a systematic framework to review changes in the relevant standards, regulations, criteria and methodologies (see NS-TAST-GD-050 – Periodic Safety Reviews).

5.40 The effects of internal and external hazards, for example those arising from dropped loads or earthquakes, may not have been addressed at the design stage for existing plants and need to be carefully considered.

5.41 Safety submissions for existing plants should contain a comparison with current standards and any significant deviation from modern design practice justified. Failure to meet modern standards should be identified by the licensee, and the implications addressed with the aim of showing that reasonably practicable improvements have been made, or will be addressed.

Loads and transients within the design basis

Design basis

5.42 The safety case should include an analysis of the potential failure modes for all conditions arising from design basis loads. The objective of the analysis is to demonstrate that the SSCs are capable of withstanding normal operating and fault loads for the projected life of the installation taking due account of potential degradation mechanisms. There should be a margin between the operating and fault envelope and the conservative failure limit over the full intended lifetime with due allowance for uncertainty. Failure modes should be progressive, with the possibility of disruptive failure without warning being remote (EDR.1, EMC.7 & EMC.11).

5.43 The safety cases for many existing structures include consideration of known or postulated degradation mechanisms or defects. Acceptance criteria based on meeting the requirements of codes and standards are not likely to be acceptable for degraded or defective structures. In some instances, for existing plants, it may be necessary to rely on ALARP arguments to enable a judgment to be made on the acceptability of safety cases. The inspector should ensure that due account has been taken of the ALARP arguments, including in the seismic analysis of the structure, and that appropriate acceptance criteria have been specified. The inspector should also ensure that the seismic safety case is compatible with the overall safety case for the installation.

Load cases

5.44 SAP FA.5 paragraph 628 requires that a safety case presents a list of all initiating faults which are included within the design basis of the plant. All loadings for operation, credible faults, accident conditions and tests should be identified and the magnitudes specified (EMC.7, EMC.11). External and internal hazard loads should also be considered as part of the design (EHA.1, EHA.3 to EHA.5 and EHA.7). Load definitions should be conservative, and remain appropriate for the future operation of the structure. This is of particular importance when reviewing proposals for extending operation or for a change of use of SSCs.

5.45 Failures of components or systems, for which acceptable case-by-case arguments have been made in accordance with SAP ERL.1 paragraph 191 do not need to be considered. It may also be appropriate to consider the resilience of structures or components to beyond design basis events.

External and internal hazards

5.46 Current standards require consideration of fault loading conditions that may not have been addressed at the design stage for existing structures. In particular, the effects on the integrity of the structure of internal and external hazards need to be addressed, EHA.1 to EHA.17 and associated paragraphs. All operational loadings and credible fault loadings should be identified and their magnitudes specified (EMC.7 & EMC.11).

Load combinations should be defined. EHA.1, EHA.3 to EHA.5 & EHA.7 cover external and internal hazard loads. Load definitions should be conservative, and remain appropriate for proposed future operation (EMC.33). This is of particular importance when assessing proposals for life extension. Further guidance can be found in NS-TAST-GD-013 and NS-TAST-GD-014 on external and internal hazards.

5.47 Failure of structures may give rise to internal hazards such as missiles, steam or hot gas release, collisions, pipe whip, which could potentially compromise other safety related structures and equipment. The safety case should demonstrate that appropriate consideration has been given to the effects of internal hazards on safety related structures, and of the secondary effects of structural failure.

5.48 External hazards can be included in the design specification for new plant and analysed in the design substantiation. However, external hazards can present difficulties for existing SSCs. For instance, an SSC may have been designed and constructed before to seismic qualification was required, or have been qualified to a less rigorous standard than that required for new structures. The position is especially challenging for existing structures whose failure would give rise to unacceptable radiological consequences, i.e. those SSCs requiring highest integrity. For advice on safety cases that claim that gross failure is so unlikely it may be discounted, see §5.13 to §5.26.

Materials: The use of proven materials

5.49 The inspector should verify that safety significant SSCs are constructed from materials with well-established properties and behaviour (EMC.13). The potential degradation mechanisms that could occur should be established at the design stage and appropriate materials chosen. Material properties used in analyses should be demonstrably conservative e.g. lower bounds of either generic databases or specific data that represent the component manufacturing and fabrication conditions. In general the steels specified in the design of pressure boundary SSCs and elsewhere have a well-established history of usage. However, if any unforeseen behaviour change or degradation mechanism is identified the licensee should review and if necessary update the relevant safety case.

5.50 The inspector should consider seeking confirmation that all metallurgical processes (including: steel-making, welding and heat treatment operations) are controlled so that steels and other materials will perform their safety functions. In addition, the effects of operational history, pressure, temperature, irradiation, creep, fatigue, and corrosion mechanisms may result in degradation in the material properties assumed at the design stage. Appropriate provision should be made for the measurement of relevant properties of representative materials (EAD.3 & EMT.6) across the full range of environmental conditions expected throughout the identified lifetime of the plant.

5.51 Material compatibility should be considered during design, before plant repairs are carried out and for maintenance activities (EMC.22 – material compatibility). For example, the possibility that materials in contact with stainless steels contain chlorides which could cause cracking should be considered.

Degradation mechanisms

5.52 The design should take account of degradation processes, including, for example, irradiation embrittlement, corrosion, erosion, creep, fatigue and ageing, and for the effects of the chemical and physical environment (EAD.1 to 5); see §5.116 to §5.125. Degradation mechanisms that may affect the SSCs should be explicitly stated in the safety case and addressed in the assessment. Failure modes should be demonstrated to be gradual and predictable (EMC.11).

5.53 The potential for interaction effects between degradation mechanisms should also be considered, e.g. creep/fatigue and stress corrosion cracking. Due allowance should be made for uncertainties in the initial state of components and the rate of degradation.

5.54 Difficulties may arise as a plant ages where particular loadings or degradation mechanisms may not have been identified at the design stage, or the understanding of the degradation mechanism changes. In these cases, it is important that the licensee's safety case considers the material performance given the modified understanding, and establishes the implications for the performance of the structure. This may involve additional examinations, sampling and testing and/or simulations of material behaviour to improve confidence in the performance of the structure. Evidence from similar plant experience elsewhere may be relevant. It may be necessary to monitor the SSC to verify that the material is not deviating from the anticipated behaviour. The inspector should examine the safety case for these aspects and look for commitments for examination and monitoring covering expected and unexpected phenomena.

5.55 A specific instance of degradation is the embrittlement of RPV steels by irradiation for which the inspector can refer to IAEA TECDOC NP-T-3.11; see Appendix A2. The adequacy of data derived from irradiation and other surveillance schemes should be examined, as appropriate, to gain assurance that they accurately represents the plant, recognising the inherent scatter in most materials properties.

Materials monitoring - the provision of in-service materials monitoring

5.56 Data derived from surveillance specimen materials may need to be examined in detail to ensure that damage mechanisms are thoroughly understood and all relevant data have been included. The appropriate use of the data in any application should be justified in the safety submission. Extrapolation might be in time or to similar base and weld materials; significant extrapolation of data should be avoided. Any extrapolation or correlation used to derive material properties should contain adequate margins to cater for uncertainties, including the effects of accelerated testing. New facilities, and where practicable existing facilities, should include surveillance material specimens and test programmes to provide adequate forewarning of detrimental material property changes throughout the life of the facility.

5.57 Test data should adequately represent the materials and conditions of interest. Materials samples might be taken from SSCs during or after manufacture or after a period of service exposure. Factors that may affect the accuracy of data are material specification, trace element content (e.g. for ferritic steel, copper in the case of irradiation embrittlement, and sulfur in the case of fatigue crack growth in some aqueous environments), heat treatment, temperature, irradiation conditions (including the thermal to fast neutron fluence ratio), environment, loading conditions and operational history. It may also be important to consider orientation of specimens with respect to the applied stress in the component.

Manufacture, testing and inspection

5.58 The starting point for metallic SSCs important to nuclear safety is compliance with the relevant design codes and specifications. There is an expectation that measures are put in place so that:

- All risks to achieving an adequate level of quality are identified and controlled
- There is evidence that each component is of adequate quality throughout its entire volume
- There is evidence that component quality is repeatable and consistent between serial components.

5.59 Material specifications, manufacturing processes and inspections should be suitable and should ensure that the SSC is free from significant defects and tolerant of any remaining defects (EMC.5, EMC.6 & ECS.3 with paragraph 169). SSCs should be designed and fabricated to facilitate examination during manufacture and service (e.g. the selection of forged rather than cast austenitic stainless steel components, to aid the transmission of ultrasound and the control of metallurgical processes to control grain size) (EMC.8 & EMC.9).

5.60 Metallurgical and other manufacturing processes, including in-process inspections, should be subject to procedural control to ensure that high standards are achieved (EMC.4). Other relevant SAPs are:

- EMC.14—Manufacture and installation should use proven techniques and approved procedures to minimize the occurrence of defects that might affect the integrity of components or structures.
- EMC.15—Materials identification, storage and issue should be closely controlled.
- EMC.16—The potential for contamination of materials during manufacture and installation should be controlled to ensure the integrity of components and structures is not compromised.

5.61 To meet high standards of structural integrity, it is necessary to establish that:

- The manufacturing processes, tests and inspections are carried out in accordance with approved procedures;
- Appropriate third party inspection of manufacture and examination is specified to ensure that a high standard of workmanship has been achieved (EMC.14 & EMC.18). Examinations of welds in highest reliability SSCs should be redundant, diverse and qualified (See Appendix A5 for guidance on qualified inspection). Pre-service inspections should be carried out at a late stage when the plant is in a state essentially as for normal operation.

5.62 Care is required in accepting commonality arguments based on manufacture, operational experience or examination of similar components. Broadly, commonality arguments are strongest where highly correlated, common cause process deviations or degradation mechanisms dominate and weakest where process deviations and degradation mechanisms have a large random element.

5.63 It is likely that the incidence of structurally significant defects will be higher than average at welds, especially those with complex combinations of material and geometry, where welding and/or access for examination or environmental conditions are difficult, and for welds for which there is no diversity of examination procedure. However, large steel forgings and castings can contain structurally significant defects; see the case studies in Appendix A3. Where a safety case requires specific assurance on the likelihood of structurally significant defects at particular locations, it can only be supported by direct examination using a technique qualified for the defect type, size and orientation of concern.

5.64 Part of an examination of the quality of manufacture should include a review of manufacturing concessions for deviations from the original specification.

5.65 Appendix A3 provides case studies related to inhomogeneities in steels that illustrate risks related to manufacturing and testing.

Welding

5.66 For new designs of SSCs, or for major modifications to existing plant, the number and location of welds should be carefully reviewed, since it may be possible to eliminate welds, to position them in areas of lower stress and lower irradiation, and ensure that they are readily inspectable (EMC.9 & EMC.10). However, the use of large forgings is not a panacea. The difficulties associated with variations of thermal transients during heat treatments and segregation tend to be more pronounced for larger forgings; see case study on the Flamanville 3 RPV domes in Appendix A3 as an example. The licensee should balance the minimisation of welds against the potential difficulties of producing large forgings with acceptable properties and inspectability throughout their volumes (EMC.5).

5.67 Appropriate documentation should be in place, prior to start of manufacture, such as an approved welding procedure specification (WPS) which details the welding parameters to be used to ensure a welded joint will achieve the specified levels of weld quality and mechanical properties. The WPS should be supported by a welding procedure qualification record (WPQR) to demonstrate that the weld will produce adequate mechanical properties, and where applicable adequate corrosion and fracture toughness properties, for the specified design conditions. The inspector should also establish that there is an adequate process in place for welders to be appropriately trained and qualified in accordance with the relevant code, standard and manufacturing procedures prior to using the WPS.

Hydrostatic testing

5.68 The specification of a hydrostatic test before service provides some assurance that the as-built SSC has been constructed to an adequate standard (SAPs paragraph 307). That is the material strength and section thicknesses are adequate. The reassurance may only be of limited value for plant where degradation mechanisms may have eroded any margins derived from the original proof tests and tests do not represent all loading conditions. Further proof tests in service are not usually feasible given the radiological consequences if failure occurred during such a test. It may also introduce additional damage to the plant in the form of stable tearing at pre-existing crack-like defects that may undermine the proof test argument.

Existing plants

5.69 It may not be possible to verify—to the same extent as new plant—that adequate standards of manufacture have been achieved. However, it should be possible to identify the manufacturer and confirm that it is, or was, a recognised company in the field. It may be possible for the licensee to examine the manufacturing records still available, and reach some conclusions on the quality of manufacture. This could reveal strengths as well as weaknesses.

5.70 When considering modifications, new SSCs should be designed, manufactured, inspected and tested in accordance with modern standards and practice where appropriate. This requires some judgment since we are dealing with what is reasonably practicable, and consistent with the overall system integrity. The inspector should refer to the TAG on ALARP, NS-TAST-GD-005).

Inspection: Manufacturing, pre-service and in-service examination and in-service monitoring

5.71 Manufacture and installation should be subject to appropriate third-party independent inspection to confirm that processes and procedures are being followed (EMC.18). In general, inspection requirements should be identified in the safety case and be incorporated into the Maintenance Schedule if appropriate.

5.72 ONR's expectations for the use of inspection qualification in relation to NDT are outlined in Reference A5. The inspector should also consult NS-TAST-GD-009 (Examination, inspection, maintenance and testing of items important to safety).

5.73 Inspection provides an important element in establishing the integrity of highest reliability SSCs. In particular, it should be demonstrated that SSCs are examined to appropriate standards (ECS.3); are as defect free as possible, with critical crack sizes at end of life being larger than the capability of the examination technique; and that the existence of defects can be established by examination throughout the operational life (EMC.5 & EMC.6). The expectation is that, for the manufacture of highest reliability SSCs, there will be a need for additional objective-based inspections beyond those necessary to achieve code-compliance; see [9] paragraph 605.

5.74 Examination immediately prior to and during service and in-service monitoring have three objectives:

- To confirm the plant is in the configuration assumed in the safety case
- To confirm any predicted degradation or ageing effect is developing within the rate allowed for in the safety case
- To confirm there is no manufacturing shortfall or degradation during storage, outages or service that is not dealt with in the safety case.

5.75 In-service examinations should be carried out where they are reasonably practicable to enable the present condition of the SSC to be confirmed, and to verify that the SSC is behaving as the safety case assumes. In-service examination provides a means of assuring that SSCs remain at all times fit for purpose (EMC.27 & EMC.28). It is noted that particular difficulties have arisen in the past in interpreting re-examination results where modifications have been made to the examination procedures following the original examinations.

5.76 For highest integrity SSCs, manufacturing, pre-service and in-service inspections should be redundant and diverse, e.g. radiography, ultrasonics, and aided surface examinations (such as liquid penetrant or magnetic particle); and possibly redundant and diverse within one method e.g. ultrasonics (EMC. 27 through EMC.30 and SAPs paragraphs 308 & 309). Where appropriate, repeat examinations should be carried out by different examination teams. The adequacy of examination procedures and personnel should be qualified. The interpretation of examination results and the assessment of their structural integrity significance should be carried out by a suitably qualified and experienced person (SQEP). For crack-like defects, the defect sizes and orientation used in integrity analyses should be pessimistic and include the contribution associated with the uncertainties in defect location and sizing for the particular examination technique. The appropriate level of pessimism in the integrity analysis will depend on the overall safety case and the consequences of failure.

5.77 The extent and periodicity of the examination proposals should be commensurate with the operational duty and safety functional requirement (EMT.6 and paragraph 209). Where defects, degradation or deviations from design intent are found in existing SSCs any proposed remedial action or technical justification should be assessed via the licensee's plant modification procedure, including Independent Nuclear Safety Assessment. Planning the extent of in-service examination based on operational experience (see §5.133 to §5.136) may be reasonable, but is not a guarantee of locating all in-service degradation in any particular plant. In general, there should be some 'speculative' element to in-service examination to look for the unexpected. A good number of degradation phenomena have been found initially by simple visual examination methods, rather than sophisticated volumetric examination techniques.

5.78 Examination results should be interpreted within an established framework of defect categorisation and sentencing criteria.

5.79 Wherever possible designs should be optimised to promote the effectiveness of inspection performed during manufacture and in-service i.e. there should be active consideration of design for inspectability. However, any inspection shortfalls should be clearly identified. For example, it may not be possible to inspect 100% of a weld because of access difficulties. The implications of the inability to inspect areas of welds should be addressed in the assessment of the significance of any defects found or defects that could exist in the areas which are difficult to access for examination.

5.80 SSCs should be designed and specified so that failure modes are progressive and sufficient warning of impending failure is provided to enable remedial measures to be taken to prevent failure or to mitigate its consequences. Monitoring may take the form of visual examination, photographic or video records, thickness measurements, or other forms of non-destructive examination (NDE) e.g. ultrasonics, eddy current, magnetic particle inspection (MPI) etc., so that degradation of SSCs can be identified before structural integrity is compromised. Monitoring should be performed at appropriate intervals to ensure that the results will enable timely identification of degradation. The inspector may also need to establish that the licensee has adequate arrangements for defining reporting and acceptance criteria, and for the evaluation of inspection and monitoring results.

5.81 The integrity of SSCs may be supported by periodic leak testing, proof testing, functional testing, strain, displacement or vibration monitoring. For existing SSCs, the inspector should consider the viability of monitoring for the remaining life of the SSC using experience of similar plant, accelerated testing, destructive testing of samples or experience in other industries, but in similar environments.

5.82 The design, manufacture, operation and maintenance of monitoring systems should be commensurate with the required duty and reliability.

Stress analysis

5.83 The design should be supported by stress analyses to demonstrate that adequate margins against failure are maintained throughout the plant life (EMC.32). The objective of the stress analysis is to demonstrate that the structure is capable to withstand normal operating conditions and fault loads for the projected life of the installation.

5.84 The material properties assumed in the stress analysis should be consistent with the requirements of the relevant design code where appropriate. Design codes implicitly include safety margins in the design assessment and assume that manufacturing has met the minimum requirements of the code. Where relevant, dynamic effects should be taken into account in the stress analysis. Where cracking is concerned, strain hardening should be considered as dynamic loads result in material embrittlement.

5.85 The analysis should take due account of potential degradation mechanisms (e.g. corrosion, creep, creep/fatigue, etc.). A requirement throughout the assessment process is that analytical models should use methods that have been verified and validated. The stress analysis should reflect the real physical condition of the plant components, including ageing and degradation (see §5.116 to §5.125). The adequacy of margins needs to be considered in the light of the perceived accuracy, reliability and conservatism of analysis and test results; see NS-TAST-GD-042 for more specific guidance on margins and uncertainties.

5.86 The data used in analyses and acceptance criteria should be clearly conservative, taking account of uncertainties in the data and their contribution to the safety case (EMC.33). Stress analysis might include finite element stress analysis (EMC.32) and, where failure by crack growth is concerned, a fracture mechanics and a fatigue assessment in accordance with recognised procedures and standards; see EMC.34 (defect sizes).

5.87 The inspector should ensure that stress analysis codes and procedures are adequately verified and validated for the particular application (EMC.32, EMC.34 and SAPs paragraph 316). The inspector can refer to NS-TAST-GD-042 (Validation of computer codes and calculation methods) for further advice.

Avoidance of fracture demonstrations

Introduction

5.88 In most cases where failure is discounted from the design basis, it means no physical defence in depth can be introduced to eliminate, mitigate or protect against the consequences of failure. Instead, conceptual defence in depth is considered, with multiple robust safety case arguments expected. While design code compliance can provide a certain amount of assurance, there are certain areas which, typically, are expected to be further reinforced in the safety case, namely: fracture analyses, reliable and readily qualified manufacturing inspections, along with conservative and achievable material properties.

5.89 To achieve this aim, a key expectation informed by precedent in the UK, relates to the integration of defect tolerance assessment, qualified inspection and conservative material properties; see Appendices A4 and A5. This is referred to as an ‘avoidance of fracture demonstration’.

5.90 Avoidance of fracture should be demonstrated for any actual or postulated degradation or defects that may remain after manufacture or that may develop during service (SAP EMC.5). Means of inspection during manufacture and throughout the full lifetime of the facility should be available to establish the existence of defects of concern (SAP EMC.6).

Defect tolerance assessments

5.91 Defect tolerance assessments combine stress analyses and materials data to estimate critical defect sizes (see Appendix A4). The ratio of the defect size that can be detected (and thus rejected) with highly reliability can be compared with the critical defect size (EMC.34), is the defect size margin (DSM) which provides an intelligible indication of the risk of fracture. In general, a safety case for an SSC where the dominant failure mechanism is fracture should not rely entirely on a fracture mechanics assessment; it should be supported by, such measures as [9]:

- Material selection and specification with tightened control of composition;
- Proven, well understood and approved manufacturing processes;
- Proven materials supported by direct fracture toughness testing of representative materials in manufacture and through life via a material surveillance strategy for SSCs that may be affected by the environment, e.g. irradiation embrittlement;
- High reliability NDT performed during manufacture; see §5.71 to §5.82;
- Where appropriate, high reliability in-service NDT; see Appendix A5 (inspection qualification in relation to NDT);
- Design for inspectability, wherever possible, designs should promote the effectiveness of the NDT performed during manufacture and in-service;

- A basis for confidence that the design intent is achievable and that there is provision for an intelligent customer capability during licensing with arrangements for third party inspection surveillance of the design and manufacturing activities.

5.92 Clearly, a DSM must be more than one. SAP EMC.28 states that an adequate margin should exist between the nature of defects of concern and the capability of the examination to detect and characterize a defect. In addition, SAP EMC.33 indicates that the data used in analyses and acceptance criteria should be clearly conservative, taking account of uncertainties in the data and their contribution to the safety case. Precedent which takes account of custom and practice, in particular the recommendations of studies of the integrity of pressurised water reactor (PWR) vessels to inform the Sizewell B public inquiry has led to acceptance of a target DSM of more than 2.0 for highest reliability components at end of life, with allowances for crack growth and degradation of material properties, in particular, fracture toughness [9, 10].

5.93 It may be appropriate to predict DSMs on initiation fracture toughness for normal operating and frequent fault conditions. However, for infrequent fault conditions, it may be appropriate to predict DSMs using less rigorous requirements. In such cases, the safety case should provide a suitable justification for any relaxation. An example of such alleviation is the use of fracture toughness data enhanced by limited stable tearing; in this case, the fracture toughness values would need to be supported by valid test data up to at least the extent of tearing invoked in the safety case (SAP EMC.33, EMC.34 and paragraph 318).

5.94 The adequacy of a DSM should be judged taking account of the overall safety case with consideration of the various factors that affect it (e.g. inspection findings, stress, materials properties, ageing and degradation) to reach a conclusion on its adequacy taking due note of previous assessment findings.

5.95 Failure modes should be progressive and the possibility of disruptive failure without warning should be remote (EDR.1, EMC.7 & EMC.11). The margin to failure should be commensurate with the consequences of failure.

5.96 The inspector should establish that the data used in the analysis are conservative, and that appropriate studies are carried out to establish the sensitivity to the analysis parameters. This aspect is especially important where a fitness for purpose analysis is concerned. DTAs should take account of ageing and degradation in service; see §5.116 to §5.125.

Fracture toughness

5.97 SAPs EMC.12 and EMC.23 require that the operating regime ensures that metal pressure boundaries exhibit ductile behaviour when significantly stressed. For other operating conditions, an RPV should be on the upper shelf wherever possible. The inspector should look for evidence that the licensee has considered all reasonably practicable measures to maximize the margin between onset of upper shelf and normal steady state operation. There are various ways of defining the onset of upper shelf conditions from a given set of materials data. The inspector should be aware that this is a complex area. An inspector who is not a specialist in this area should seek informed advice.

5.98 Ferritic steel components should be operated on the upper shelf of fracture toughness as far as possible under all potential operating and fault conditions. Situations where this target might be relaxed include expected, practically unavoidable but short duration loads (e.g. certain phases of start-up and shut-down) or low frequency fault conditions. Such situations need to be carefully justified by the safety case. Where upper shelf conditions cannot be achieved, it is important that all uncertainties are considered and that adequate margins on toughness are shown. For existing SSCs, it may be possible to alleviate concerns about low temperature operation by introducing limiting temperatures for operation. For start-up and shut-down situations, pressure-temperature limit diagrams are likely to be required. For new plant, and where practicable for existing plant, there is a preference for safety relief devices with set-points under the control of the protection system, to provide automated compliance with the pressure-temperature limit diagram.

Fracture toughness Master curve

5.99 The fracture toughness Master curve is a representation of the cleavage fracture toughness behaviour of low alloy RPV steels and weld metals [11]. The fracture toughness Master curve uses fracture mechanics theory to predict the effect of thickness on, and the variability of, cleavage fracture toughness. Curve fitting is used to predict median values of fracture toughness versus temperature.

5.100 The UK technical advisory group on structural integrity (TAGSI) has considered how the Master curve can be used to predict the behaviour of full scale structures from tests on small specimens. While the TAGSI welcomed the Master curve approach [12], it gave cautionary restrictions on its use and guidance to ensure that its application to structural integrity assessments of nuclear plant was soundly based. These related to:

- **The thickness correction** — The thickness correction for the Master curve followed from accepted assumptions for a stress field ahead of a sharp crack under small scale yielding; however, if small scale yielding was not retained, the assumptions would be expected to break down. On this basis, the correction would not be strictly valid at higher values of fracture toughness where the plastic zone was not contained or where there was a loss of crack tip constraint. Also, extensive research into size effects suggested that different values of the thickness correction exponent were appropriate over different parts of the toughness transition curve, particularly when ductile tearing preceded cleavage fracture; this was not addressed by the Master curve procedure.
- **Its application to long structural cracks** — The TAGSI considered that this was unrealistic and unduly conservative for application to long structural cracks. It recommended that, where the crack front was longer than the thickness of a component being assessed, the reduction should use a correction based on the ratio of the reference thickness to the actual material thickness.
- **The effects of ductile tearing** — The theoretical basis of the Master curve is consistent with cleavage fracture not ductile tearing. While the experimental validation of the Master curve appeared to support acceptance of some ductile tearing, extensive statistical analysis of fracture toughness data in the UK indicates that different coefficients to the transition curve equation may be appropriate for failure by ductile tearing and by cleavage fracture. Therefore, the TAGSI recommended that use of the Master curve equations should be limited to ductile tearing not exceeding the smaller of 0.2 mm and 5% of the length of the ligament from the crack tip to the back face unless validated by experiment.
- **The upper validity upper limit on fracture toughness** — The TAGSI was concerned that the upper validity limit on fracture toughness in the ASTM standard on Master curve (E1921) was not sufficiently restrictive to ensure that there was only small scale yielding with nil or very limited stable crack growth.
- **The number of specimens tested and data pooling** — ASTM E1921 allows the use of as few as six specimens to determine the reference temperature (T_0). However, the precision of an estimate of T_0 is sensitive to scatter and

more specimens might be needed. Also, rather than averaging separate estimates of T_0 from each temperature, TAGSI recommended pooling the data and using a maximum likelihood approach to make efficient use of the data.

5.101 Overall, ONR has accepted the use of Master curve in cases where it is experimentally validated. The validation for any future use related to nuclear safety would, as a minimum, be expected to provide evidence to address the recommendations and issues above identified by the TAGSI [12].

Inferences from hydrostatic tests

5.102 Arguments might be used to show that defects which could have survived a hydrostatic (proof) test would not grow in service such that they could threaten structural integrity at the end of life under the most onerous loading condition. Such arguments may need to be viewed with some caution since the original margins may be eroded by service conditions and time-dependent degradation mechanisms. The test may also not represent the most onerous crack tip loading situation in-service. The possibility also exists that ductile tearing and/or deformation near a crack tip may have occurred during the proof test. In addition, it should be emphasised that design and assessment codes such as R6 are failure avoidance analysis techniques and are not primarily intended as methods for failure prediction. The primary (and historical) purpose of pressure testing is to confirm the adequacy of material strength, wall thicknesses and mechanical closure arrangements. At present, it is not accepted that adequate validation has been completed to enable ONR to have high confidence in proof test analyses for the avoidance of fracture.

Leak detection and leak before break

General concept

5.103 The leak before break concept (LBB) was originally developed in the US as a means of justifying the elimination of the dynamic effects of postulated high energy pipe ruptures from the design basis of nuclear power plants. LBB refers to a situation where it is argued that a defect, were it to grow, would lead to a leak rather than a break. In practice, the application of LBB is dependent on confidence that defects on break-through, and over time, will remain stable and detectable. The application of the LBB concept also strongly depends on the location and the capabilities of the leak detection system.

5.104 ONR policy is that good design practice would be followed to ensure defence in depth in the plant design (EKP.3). This is informed by a rigorous consideration of the consequences (direct and indirect) of postulated failure with gross failure usually limiting. The use of LBB to argue partial failure or in effect to discount gross failure is not consistent with this approach and an inspector would need to consider the circumstances in which it was being claimed. In addition, there are a number of uncertainties in the assumptions in LBB assessments (e.g. crack opening area, friction coefficient, etc.) that make it difficult to demonstrate that LBB will occur in practice.

Applicability of the leak before break concept for high reliability components

5.105 Where high reliability is claimed, an LBB argument will not be appropriate as the main safety case argument since a high reliability claim would not be consistent with the concept of LBB. However, LBB is recognised as a supporting argument which can be used to demonstrate defence in depth.

5.106 Break Preclusion is a deterministic concept, which ensures by preventative measures (requirements on design, materials, product forms, manufacturing, quality assurance, analysis of fatigue crack growth) and surveillance measures (requirements on transient and water chemistry monitoring, leakage detection, in-service inspection) that rupture of a pipe can be discounted in safety studies. The use of break preclusion may have provisions which support the achievement of highest reliability, but further measures are usually necessary to meet UK expectations if the safety case discounts gross failure.

LBB demonstration

5.107 Leak detection and LBB arguments might be provided to support pressure boundary structural integrity safety cases. In such cases, leak detection capability is fundamental (EMC.25, EMC.26 and SAPs paragraph 312). The safety case should explain the leak detection system and identify the sensitivity, the reliability, the response time and availability of the leak detection system. There is likely to be a need for periodic testing and calibration of leak detection equipment.

5.108 Claims in the safety case must be consistent with the practicalities of the leak detection system. The inspector should examine the safety case for operating instructions covering how operations staff should respond to the detection of a leak. The response may be graded depending on the rate of leakage and the rate of change of leakage.

5.109 The hazards associated with the leakage of fluids should be considered in the safety case to ensure that they do not lead to potential loss of safety related plant or equipment and do not pose a hazard to operators. The safety case should demonstrate that the plant can continue to be operated safely in the event of leakage or spillage of fluids.

5.110 For an LBB argument, the degradation mechanisms of the components should be evaluated in the safety case. The LBB approach is difficult to apply to piping or SSCs that can fail in service from unanticipated loads or active degradation mechanisms e.g. water hammer, creep, erosion, corrosion or excessive fatigue. Such effects would result in defect or loading conditions different from those postulated during the LBB assessment and could therefore invalidate the LBB argument. Guidance is given in IAEA-TECDOC-710 - Applicability of the leak before break concept; see Appendix A2.

5.111 Plant inspections are preferred to LBB as a means of monitoring the plant condition. Such information can be used to demonstrate forewarning of failure and the continuing validity of the assumptions in the safety case e.g. the absence of defects of concern, degradation mechanisms and the understanding of the loadings/environments.

5.112 From operational experience (see §5.133 to §5.136), incidents of sub-critical crack growth in boiling water reactor (BWR) and PWR primary circuit and connected system piping have to date resulted in stable through-wall cracks which have leaked and remained stable until the leak has been detected. In the recorded incidents, most of the pipework has been made from austenitic stainless steel. By one means or another and eventually, the leakage has been detected. The relevant incidents have involved normal plant loadings (which applied during the sub-critical crack growth). That is, the through wall cracks were not subjected to a fault loading. Large scale experiments on ferritic and stainless steel pipework sections show the resilience of nuclear plant type pipework (circa 350 mm outside diameter and 25 mm wall thickness) to large, dynamic, repeating ‘fault’ type loadings, combined with normal pressure and temperature [13, 14]. However, this operational experience does not amount to a safety case.

5.113 In general, it would be expected that a leak detection or leak-before-break argument would be more easily made and accepted for thin-walled components, made from ductile materials. Operational experience data (e.g. [15]) predominantly contains leak type failures in small diameter, thin wall pipework. This may be due to the relative lack of attention to design and in-service conditions for ‘minor’ lines of ‘low’ perceived safety significance, rather than an inherent propensity for small diameter, thin wall pipe to leak or burst compared to large diameter, thick wall pipe. If a thin-walled component also has a small diameter, a leak from a through-wall defect may be difficult to detect because of the absolute length and gape of the defect. The inspector might decide to place little weight on a leak detection / leak-before-break argument for a thick-walled component or where the limiting through-wall crack length is only a small multiple of the wall thickness.

5.114 The inspector should consider whether the analysis assumptions are consistent with the overall fatigue and fracture analysis, and that a sufficient margin is available between the capability of the leak detection system to detect a leak and failure of the component. It is important to ensure that the component is operating in a ductile state of fracture toughness where leak-before-break is claimed. Clearly, if a through wall crack is postulated to be detectable by the leakage through the crack during normal operation, the defect needs to be stable with a suitable margin under the range of normal operation loading conditions. Margins for the through-wall defect under infrequent fault loads are a separate matter. Factors that may need to be considered include the potential for debris blockage of a leak path, the dynamic effects at break-through of the crack to a through-wall crack and possibly initial break-through over only a fraction of the complete crack length.

5.115 LBB arguments might not be applicable if interacting, multiple defects, rather than isolated defects, are possible. The examination history may give an indication of the likelihood of such defects. The inspector should be aware that, on occasion, defect indications appear not to have been reported even when the manufacturing inspection procedures appear to have been suitable and sufficiently sensitive (Appendix A4).

Ageing and degradation

5.116 In preceding sections of this TAG, aspects of ageing and degradation have been implicitly considered. For metal SSCs, the consideration of ageing and degradation at the design stage and during service is long established practice. Mechanisms such as creep, fatigue, thermal strain ageing, irradiation embrittlement, environmental effects such as corrosion and flow assisted corrosion, are well recognised and routinely considered in structural integrity evaluations. Ageing and degradation mechanisms might lead to initiation of defects, sub-critical growth of pre-existing defects or reduce the defect tolerance of the material. Some ageing and degradation mechanisms might lead to some or all of these effects.

5.117 Monitoring and surveillance should be appropriate for the rate of progress of anticipated degradation mechanisms as well as giving some speculative coverage for unexpected degradation processes (EAD.1 to EAD.4).

5.118 Within the context of nuclear regulation in the UK, and unless there is an acute problem of ageing or degradation, the natural stage to consider ageing, degradation and obsolescence issues is during the Periodic Safety Review process.

5.119 In assessing the structural aspects of corrosion and materials degradation threats, there is an important interface between ONR’s chemistry and structural integrity specialisms. ONR TAGs NS-TAST-GD-088 (Chemistry of operating civil nuclear reactors) and NS-TAST-GD-089 (Chemistry assessment) are relevant to structural integrity. The inspector should be aware of the guidance contained within these TAGs, e.g. section on corrosion and materials degradation (Section 6.2 of NS-TAST-GD-089).

5.120 Of particular importance are degradation mechanisms in SSCs that are difficult or impractical to inspect in service. In these cases, conservative estimates of the minimum safe working life of the SSCs should be included in the design and appropriate surveillance schemes specified. For SSCs that are impractical to replace, the conservative estimate of minimum safe working life should be especially robust. For SSCs that cannot be replaced, the use of novel materials or design concepts is unlikely to assist in establishing a conservative safe working life (EAD.1). The safety case for nuclear facility SSCs needs to include a suitable allowance for the effects of ageing and degradation on the safety margins throughout plant life, including decommissioning. SSCs should be designed to be tested, maintained, repaired and inspected or monitored periodically in terms of integrity and functional capability over the lifetime of the plant, without undue risk to workers and significant reduction in system availability. Where such provisions cannot be attained, proven alternative or indirect methods should be specified and adequate safety precautions taken to compensate for potential undiscovered failures. Operational histories should be available for the lifetime of the plant.

5.121 It is to be expected there will be uncertainties in material properties and plant parameters required in the estimation of a safe working life (EAD.2). Such uncertainties should be considered during the design process and subsequently confirmed or otherwise by in-service monitoring and measurement of material properties and plant parameters.

5.122 Periodic review during service should use evolving in-service information to update the predicted minimum safe working life (EAD.3). Ageing and degradation mechanisms have the potential to erode safety margins attributed to the plant at start of life. Clearly, this can have safety significance. Therefore each nuclear facility should have an ageing management programme. The purpose of these programmes is to monitor changes in the relevant materials properties such as fracture toughness in the RPV beltline regions for example. Uncertainties in the measurements and material variability should be taken into account in the assessment.

5.123 Where parameters relevant to the design of plant could change with time and affect safety, provision should be made for their periodic measurement (EAD.4). The actual operation conditions should be within the design envelope of the SSCs. The means and frequency of SSC monitoring should be consistent with the degradation mechanism in question (EMC.26). The extent and periodicity of the examination proposals should be commensurate with the SSCs operational duty and associated safety functional requirements. Provisions for non-routine inspection following extreme events or faults that could accelerate degradation mechanisms should be made. Uncertainties and limitations in any inspection should be explicitly reported. Parameters to be monitored or inspected should be clearly defined along with the data assessment methods, relevant acceptance criteria and any corrective actions that may be necessary, particularly at the start of in-service operation. Provisions for on-line monitoring should be considered, particularly when this would provide forewarning of degradation leading to the failure of SSCs and when the consequences of failure could be important to safety.

5.124 SAPs paragraph 31 & 33 [2] on lifecycle and on facilities built to earlier standards are relevant to existing SSCs where, by comparison with modern standards, shortcomings may be present and it may not be possible to introduce changes. In these cases, measures such as changes to operating conditions may be necessary and/or consideration should be given to the reasonable practicability of enhancing confidence in the safety case by additional research, examination, measurements, material examination, analysis, or enhanced monitoring or make alternative provisions to ensure safety. Clearly, if adequate arrangements cannot be made, it will be necessary to a terminate operations at a facility.

5.125 Obsolescence is the non-physical ageing of SSCs, i.e. the process of their becoming out of date owing to the availability and evolution of knowledge and technology, suppliers, SQEP and the associated changes in requirements, codes and standards. Existing SSCs may have been designed and built to a code or standard that is no longer current (i.e. obsolete) which usually will be superseded by a current code or standard. Current relevant codes and standards can form the basis of a design capability assessment; see EAD.5.

Quality assurance (QA)

5.126 There should be appropriate quality assurance throughout all stages of design, procurement, manufacture, installation, commissioning, operation and decommissioning. Quality assurance arrangements are also required for production of the safety case. The specification and supply chain management of SSCs should also be the subject of periodic review and quality assurance oversight. For further advice, the inspector can refer to NS-TAST-GD-077 (Supply chain management arrangements for the procurement of nuclear safety related items or services).

5.127 The licensee should use, and require its contractors to use, formal QA procedures to specify the quality and organisational arrangements for each stage of design, manufacture, construction, installation, commissioning, operation and decommissioning. The QA Programme/Management arrangements should be sufficient to support the claims of the safety case. The QA Programme/Management arrangements should comply with recognised standards and where appropriate should include provision for the appointment of an Independent Third Party Inspection Agent (ITPIA). The aim should be to provide confidence that the safety case requirements have been met by control and surveillance of the design, manufacture, inspection, testing, operation and maintenance activities.

5.128 From experience of where issues can arise, the inspector may wish to check the licensee-contractor interface and other organisation-to-organisation interfaces of the QA arrangements. The licensee's supply chain arrangements for products should include technical awareness and not just be a procurement and financial process. The licensee should have a process for checking (perhaps on a sampling basis) the veracity of 'certificates' for products, especially where the products might be considered 'commodity' items and have been through a chain of suppliers before ultimate delivery.

Non-conformances

5.129 The QA arrangements should include a procedure for dealing with non-conformances so that departures from design, specification of materials, manufacturing processes, dimensional tolerances, defects etc., can be identified and appropriate consideration given to their safety significance. When appropriate, this procedure may result in concessions allowed by the Design Authority against the original design intent or requirement. It should be demonstrated and recorded that the SSC is capable of meeting its safety functional requirements, if necessary, by remedial work (EMC.19). The range of technical disciplines involved in reaching judgments on non-conformities and concessions should be appropriate to the issues involved. To provide confidence in the quality of the design, manufacture, examination and testing, the inspector should consider examining the system for dealing with non-conformances on a sample basis. A review of the case history or lifetime records (the terms vary among licensees) may be appropriate during manufacture of new SSCs, during periodic reviews or discovery of unexpected defects in existing SSCs. The aim is to verify that any concessions granted do not invalidate the safety case requirements or assumptions.

Lifetime records

5.130 Lifetime records of the manufacture should be comprehensive, indicating inter alia the identities of the heat and quality plan, elemental analysis data, mechanical test data, the results of non-destructive examinations and the metallurgical processes (EMC.20).

5.131 The inspector may need to examine the lifetime records (case histories) to verify that they contain the detailed weld design, weld procedures, welder qualification, and weld inspection procedures. Examination of lifetime records can provide confidence in the original manufacturing quality. This is of particular importance in terms of the original weld inspection procedures and results. Nevertheless, original construction records do not always show the full picture (e.g. weld repairs may not have been recorded accurately or examination records may be incomplete) and the inspector may need to examine whether the licensee has considered other options, such as a re-examination if reasonably practicable.

5.132 For complex, multi-disciplinary safety cases the inspector may wish to consider communication of information between disciplines and the handling of issues generated during the production of the safety case; SAPs MS.2, MS.4 & EHF.8 and paragraphs 58, 60 to 68, 77, 98 and 457, as they relate to production of safety cases; and the link between the assumptions and claims made by the safety case and the evidence of the plant condition and operation).

Operational experience

5.133 Wherever possible, the design and operation of metal SSCs (and associated safety cases) should be informed by relevant specific and general operational experience (SC.7 and paragraphs 99 and 100). Similarly, the inspector's assessment of a safety case should take into account relevant operational experience. However, total worldwide experience of nuclear reactors is modest; for example, to end 2015, worldwide operational experience for water-cooled reactors was about 15 000 reactor years and operating experience for the UK advanced gas-cooled reactors (AGRs) a few hundred reactor years. Claims based on operating experience should reflect this, particularly for low likelihood events.

5.134 In general, detailed information on operational experience is proprietary and not freely available. However, ONR staff can access the IAEA Incident Reporting System (IRS) which provides information on nuclear power plants. Users of the IRS should be aware that it does not include all events; it contains the most significant events and in most cases includes the range of causes of failure. This is clear from comparisons between the numbers of incidents reported by different sources, e.g. [15 & 16]. The inspector should ensure that any claims based on information from the IRS reflect the extent of its coverage and, when appropriate, liaise with other national regulators.

5.135 In assessing a safety case which includes operational experience, the inspector should review the weight of the operational experience and its role in the safety case. Relevant questions related to an operational experience database are:

- What are the least frequent failure sequences that are included?
- How many examples of frequent fault sequences are included?
- What is the size of the database in terms of plant-operating years?
- What is the event collection process?
- What is the relevance of the experience to the safety case?

5.136 It is important to consider available operating experience in the production and assessment of safety cases. But available operating experience alone is unlikely to be an adequate basis for a safety case.

Decommissioning

5.137 Approaching and during the decommissioning of a nuclear facility, the continuing, amended or reducing role of the integrity of metal SSCs should be reviewed. Changes to the role of metal SSCs for the decommissioning phase should be incorporated in the decommissioning safety case.

5.138 Several of the factors for normal operation of a nuclear facility (e.g. in-service examination, ageing and degradation, materials monitoring) may remain relevant during decommissioning. However, the requirements may be less demanding for the decommissioning phase compared with normal operation; it depends on the residual nuclear hazard. The inspector should apply the guidance in this TAG to the decommissioning phase of a nuclear facility, moderated by the changing nuclear hazard. A basic factor is whether, under generally accepted definitions, what were defined as pressure systems for normal operation, continue to be classified as pressure systems in the decommissioning phases. Further advice on decommissioning is available in TAG-026 (decommissioning).

6. REFERENCES

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[2] ONR, Safety Assessment Principles for Nuclear Facilities, 2014 Edition, Rev. 0 (www.onr.org.uk/saps/saps2014.pdf)

[3] ONR, 'Enforcement' [management model], ONR-ENF-GD-006, Rev. 2, 2019 (CM9 2019/205459)

news.onr.org.uk/2018/04/new-onr-enforcement-management-model/

[4] ONR, Enforcement policy statement, April 2019 (www.onr.org.uk/documents/enforcement-policy-statement.pdf)

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www.onr.org.uk/new-reactors/reports/onr-gda-007.pdf

[10] ONR, 'Step 4 structural integrity assessment of the EDF and AREVA UK EPR™ reactor,' ONR-GDA-AR-11-027, Rev. 0, 2011 (CM9 2010/581504)

[11] ONR File note 'The fracture toughness Master curve,' December 2018 (CM9 2019/206776)

[12] TAGSI, 'Comments by TAGSI on the ASTM standard E1921-98 the Master curve method for establishing fracture toughness curves,' P(00)164, 2000 (CM9 2018/333169)

[13] HSE, 'Record of large scale tests on pipes, International piping integrity research - 1.3 Facility, 1.3 Tests, 1.1/1.2 Tests summary,' (Video) (HSE Library Item R132 985)

[14] HSE 'Pipe system test on aged cast stainless steel: Experiment and slow motion: IPIRG Test 1.3-7, Battelle,' (Video) 1990 (HSE Library item R133 509)

[15] Harrop L P, 'Tables summarizing IAEA Incident Reporting System (IRS) Incident Reports for nuclear power plant piping degradation - cracks, leaks, ruptures in PWR and BWR reactor coolant loops and connected system piping,' 2007 (CM9 2007/341485)

[16] Heurta A, 'OECD NEA Related Project SCAP Project, OPDE, RI-ISI and RISMET. OECD IAGE Sub-group on integrity of metal components and structures,' (Presentation), 2007 (CM9 2008/128)

7. GLOSSARY AND ABBREVIATIONS

AFNOR	Association française de normalisation
AGR	Advanced gas-cooled reactor
ALARP	As low as reasonably practicable
ASN	Autorité de sûreté nucléaire (Nuclear safety regulator - France)
BSL	Basic safety level
BSO	Basic safety objective
BWR	Boiling water reactor
DSRL	Decommissioning safety reference level (WENRA)
DSM	Defect Size Margin
DTA	Defect tolerance assessment
EMM	Enforcement management model
ENIQ	European network for inspection qualification
ESPN	Arrêté du 12 décembre 2005 relatif aux équipements sous pression nucléaires (French law on NPE, the ESPN Order)
FANC	Federaal Agentschap voor Nucleaire Controle (Nuclear safety regulator - Belgium)
FESI	Forum for engineering structural integrity
GDA	Generic design assessment
HAZ	Heat affected zone
HSE	Health & Safety Executive
IAEA	International Atomic Energy Agency
IPIRG	International Piping Integrity Research Group
ITPIA	Independent third party inspection authority
INSA	Independent nuclear safety assessment
IQ	Inspection qualification
IRS	IAEA incident reporting system
LBB	Leak before break
LC	Licence condition
MPI	Magnetic particle inspection
NDE	Non-destructive examination
NDT	Non-destructive testing
NPE	Nuclear pressure equipment
OECD	Organisation for Economic Cooperation & Development
PSA	Probabilistic safety analysis
PWR	Pressurised water reactor
QA	Quality assurance
RPV	Reactor pressure vessel
RSRL	Reactor safety reference levels (WENRA)
SAP	Safety assessment principal (ONR) [1]
SQEP	Suitably qualified & experienced person
SSC	Structures, systems & components
TAG	Technical assessment guide (ONR)

TAGSI	Technical advisory group structural integrity
WENRA	West European Nuclear Regulators Association
WPQR	Welding procedure qualification record
WPS	Welding procedure specification

8. APPENDIX A1: WENRA: REACTOR SAFETY, DECOMMISSIONING AND WASTE AND SPENT FUEL STORAGE REFERENCE LEVELS

A1.1 This appendix considers the implications of the Western European Nuclear Regulators Association Reference Levels for advice in the TAG. The publications are available from the WENRA website (www.wenra.org/publications/).

WENRA safety reference levels for existing reactors

A1.2 The scope of WENRA Reactor safety reference levels (RSRLs) report for existing reactors [A1-1] is consistent with the title. The current version (2014) addresses lessons learnt after the TEPCO Fukushima Dai-ichi nuclear accident, including insights from the EU stress tests:

- 05 Issue 5: Design basis envelope for existing reactors: E5 Set of design basis events—lists internal hazards in §E5.1 and external hazards in §E5.2.
- 05 Issue 5: Design basis envelope for existing reactors: E9 Design of safety functions—states in §E9.6:

  'The means for shutting down the reactor shall consist of at least two diverse systems'

This is not primarily a structural integrity issue, though there may be potential structural integrity failures which could compromise the physical means of shutting down a reactor.

- 11 Issue K: Maintenance, in-service inspection and functional testing: K3 Implementation—states in §K3.9:

  'The reactor coolant pressure boundary shall be subject to a system pressure test at or near the end of each major inspection interval.'

This is distinct from a leak test, which is the subject of §K3.8; the implication is that the test pressure would be above the design pressure. This practice is not adopted in all countries, including the UK.

WENRA decommissioning safety reference levels

A1.3 The WENRA Decommissioning safety reference levels (DSRLs) report [A1-2], do not explicitly contain requirements for metal SSCs. Advice on decommissioning is given in §5.137 and §5.138 of this TAG.

WENRA Waste and spent fuel storage safety reference levels

A1.4. The WENRA waste and spent fuel storage safety reference level (WSFSSRL) report [A1-3] does not explicitly contain requirements for metal components and structures. However, aspects of this TAG may be relevant to assessing the structural integrity of storage containers.

References

[A1-1] Western European Nuclear Regulators Association (WENRA), 'WENRA reactor safety reference levels,' 2014 (CM9 2015/390992) (www.wenra.org/publications/)

[A1-2] Western European Nuclear Regulators Association (WENRA), WENRA decommissioning safety reference levels, Version 2.2, 2015 (CM9 2019/363536) (www.wenra.org/publications/)

[A1-3] Western European Nuclear Regulators Association (WENRA), WENRA waste and spent fuel storage safety reference levels, Version 2.2, 2014 (CM9 2019/363516) (www.wenra.org/publications/)

9. APPENDIX A2: IAEA STANDARDS, GUIDANCE AND DOCUMENTS

A2.1 This appendix considers the implications of the IAEA documents for advice in the TAG. The documents are available on the IAEA website (www.iaea.org/publications).

The documents

A2.2 IAEA publishes several types of documents, grouped into series. The four series of interest here are:

- Safety standards series
- Safety reports series
- Technical reports series
- Technical documents series (TECDOCS).

The IAEA safety standards comprise: safety fundamentals; safety requirements; and safety guides [A2-1]. These IAEA safety standards are applied by the IAEA and joint sponsoring organisations to their own operations and are recommended for use by states and national authorities and by other international organisations in relation to their own activities. IAEA documents not listed under the safety standards series are not part of IAEA safety standards. For TECDOCs, see the IAEA disclaimer in the reference list below.

A2.3 The result of this review of IAEA documents can be summarised as:

- The review of the IAEA documents has not revealed any significant gaps in the SAPs or TAGs
- The IAEA Safety Series documents leave the inspector a good deal of latitude for judgment and do not appear to constrain the inspector to any greater extent than the SAPs and TAGs.

A2.4 The lists of IAEA documents have been reviewed and the subset that is potentially relevant to assessments of metal SSCs extracted to provide the references in this appendix.

A2.5 In addition to the search of IAEA documents with titles relevant to the assessment of the integrity of metal SSCs, a general search of IAEA documents was made using the word 'advanced' to identify IAEA documents relevant to new reactors and a number of documents were identified. These are not primarily concerned with structural integrity, but they do give an overview of new designs and the role of structural integrity within those designs. The documents found as a result of this search and filtering are listed as references in this appendix.

Safety standards series

A2.6 Safety Fundamentals (SF-1) [A2-1] is the primary publication in the IAEA safety standards series. SF-1 is a high level document; it contains ten Principles. These Principles have their equivalents in the more general Principles of the ONR SAPs. There is no SF-1 Principle that relates specifically to the assessment of the structural integrity of metal SSCs.

A2.7 Two documents are relevant to this TAG, SSR-2/1 and SSR-2/2 [A2-2 & A2-3]. According to SSR-2/1, IAEA safety requirements establish the requirements that must be met to ensure safety. These are expressed as ‘shall’ statements and are governed by the objectives and principles presented in the safety fundamentals.

A2.8 IAEA SSR-2/1 [A2-2] covers requirements for design of nuclear power plants at a broad level. The general requirements are applicable to the assessment of structural integrity of metal SSCs. There are also some requirements specific to assessment of metal SSCs. Overall, if the inspector follows the SAPs and the guidance in this TAG, the Requirements in IAEA SSR-2/1 will be addressed. Particular sections of IAEA SSR-2/1 of interest here are:

- Requirements for management of safety
- Principal technical requirements
- Requirements for plant design
- Requirements for design of plant systems
  - Reactor core and associated features
  - Reactor coolant system

A2.9 A point-by-point comparison between the IAEA SSR-2/1 [A2-2] requirements and the SAPs would be unduly cumbersome. However, it is worth noting the two following requirements:

Requirement 4: Fundamental safety functions

> *'Fulfilment of the following fundamental safety functions for a nuclear power plant shall be ensured for all plant states: (i) control of reactivity, (ii) removal of heat from the reactor and from the fuel store and (iii) confinement of radioactive material, shielding against radiation and control of planned radioactive releases, as well as limitation of accidental radioactive releases.'*

Requirement 47, §6.14

> *'The design of the reactor coolant pressure boundary shall be such that flaws are very unlikely to be initiated, and any flaws that are initiated would propagate in a regime of high resistance to unstable fracture and to rapid crack propagation, thereby permitting the timely detection of flaws.'*

A2.10 Several of the safety requirements in [A2-2] are relevant for the integrity of metal SSCs including inter alia:

- Requirement 9: Proven engineering practices
- Requirement 10: Safety assessment
- Requirement 15: Design limits
- Requirement 22: Safety classification
- Requirement 28: Operational limits and conditions for safe operation
- Requirement 31: Ageing management
- Requirement 44: Structural capability of the reactor core boundary.

This is covered by SAPs EMC.1, EMC.5, EMC.6, EMC.11, EMC.12, EMC.23, EMC.26 and EMC.34.

A2.11 SSR-2/2 [A2-3] covers general requirements for commissioning and operation. Relevant sections include:

- Management of operational safety: Operational limits and conditions (§4)
- Plant commissioning (§6)
- Plant operations (§7)
- Maintenance, testing, surveillance and inspection (§8)

A relevant example for SSR-2/2 is §4.6 which states:

> *'The operational limits and conditions shall form an important part of the basis for the authorization of the operating organization to operate the plant. The plant shall be operated within the operational limits and conditions to prevent situations arising that could lead to anticipated operational occurrences or accident conditions, and to mitigate the consequences of such events if they do*

occur. The operational limits and conditions shall be developed for ensuring that the plant is being operated in accordance with the design assumptions and intent, as well as in accordance with its licence conditions.'

This is covered in the SAPs by EMC.21, EMC.22 and EMC.23 and paragraphs 295 & 310.

A2.12 Safety guides that are relevant to this TAG include: NS-G-1.9, NS-G-1.10, NS-G-2.6, NS-G-2.3, NS-G-1.6 [A2-4 to A2-10]. According to the IAEA, its safety guides recommend actions, conditions or procedures for meeting safety requirements. Recommendations in safety guides are expressed as 'should' statements, with the implication that it is necessary to take the measures recommended or equivalent alternative measures to comply with the requirements.

A2.13 These guides are aimed at the designer and operator of the facility, not the regulator. However the inspector may find their content useful as background information.

A2.14 NS-G-1.9 [A2-4] provides guidance for the design of the reactor coolant system and associated systems. The guidance is general and specifically oriented to PWR, BWR and heavy water reactors. Sections specifically relevant to this TAG are:

- General considerations in design
  - Selection of materials
  - Provision of overpressure protection
  - Considerations of isolation
  - Provisions for in-service inspection, testing and maintenance
- Specific considerations in design
  - Reactor coolant system

An example of how NS-G-1.9 and the SAPs cover particular topics are the 'Provision of overpressure protection' in NS-G-1.9 (§3.39 to §3.46) and 'Consideration of isolation' (§3.66 to §3.69) which are covered in the SAPs in EPS.3 to EPS.5 (overpressure) and EPS.2 (isolation).

A2.15 NS-G-1.10 [A2-5] provides guidance for the design of reactor containment systems. For this TAG, the only relevance is if the containment structure includes a significant metal component, for instance an inner steel pressure shell surrounded by a concrete outer shell. The SAPs EMC.1 to EMC.34 and ECE.1 to ECE.24 cover the same ground as NS-G-1.10 in terms of structural integrity.

A2.16 NS-G-2.6 [A2-6] covers maintenance, surveillance and in-service inspection. For assessment of integrity of metal SSCs it is mainly the surveillance and in-service inspection aspects which are relevant. Sections potentially relevant to this TAG are:

- Analysis of results and feedback experience
- Area in which special considerations apply
  - Plant ageing
  - Plant designed to earlier standards
- Additional considerations specific to surveillance
- Additional considerations specific to in-service inspection.

The few paragraphs on plant ageing and plants designed to earlier standards provide general points. SAPs which address surveillance and in-service inspection for metal SSCs are EAD.1 to EAD.5, EMC.24 to EMC.30. General matters on maintenance, inspection and testing are covered in EMT.1 to EMT.8.

A2.17 NS-G-2.3 [A2-7] covers modifications in general. As defined in NS-G-2.3, modifications can include physical changes to plant, operational limits and conditions, operating procedures and modifications to safety assessment tools and processes. NS-G-2.3 does not seem to include the concept of modification to a safety case, without any of the modifications listed above. ONR might decide to assess a revision to a safety case that argued that no modification was needed to address an emergent issue or new knowledge. Active management and maintenance of safety cases, which implies the potential for their modifications, is addressed in SAP SC.7. For the integrity of metal SSCs, ONR SAP EMC.31 applies. There is no separate section of the SAPs which covers modifications. It may be that — as ONR expects modifications to be covered by safety case changes, at the appropriate categorisation of the safety function of the SSC — the SAPs deal implicitly with modifications through the principles relevant to safety cases (SC.1 to SC.8).

A2.18 NS-G-1.6 [A2-8] provides guidance on seismic design. Relevant sections are:

- Seismic design
  - Piping and equipment
- Qualification by analysis
- Seismic qualification by means of testing, earthquake experience and
  - Indirect methods

Seismic loading is an external hazard covered by SAPs EHA.1 through EHA.17. In terms of the integrity of metal SSCs, seismic loading is included (implicitly) in EMC.7.

Safety reports series

A2.19 The remaining IAEA documents in the reference list below are in the technical report series and the TECDOC series. These 'informational publications' do not contain IAEA principles, requirements or guidance. The technical report series and TECDOC Series documents listed here are included because their titles appear relevant to the integrity of metal SSCs.

References

Safety standards series

A selection of relevant documents from the documents listed on the IAEA website:

[A2-1] IAEA safety standards series: Fundamental safety principles: Safety fundamentals SF-1, 2006
[A2-2] IAEA safety standards series: Safety of nuclear power plants: Design: Specific safety requirements SSR-2/1, Revision 1, 2016
[A2-3] IAEA safety standards series: Safety of nuclear power plants: Commissioning and operation for protecting people and the environment: Specific safety requirements SSR-2/2, Revision 1, 2016
[A2-4] IAEA safety standards series: Design of the reactor coolant system and associated systems in nuclear power plants: Safety guide NS-G-1.9, 2004
[A2-5] IAEA safety standards series: Design of reactor containment systems for nuclear power plants: Safety guide NS-G-1.10, 2004
[A2-6] IAEA safety standards series: Maintenance, surveillance and in-service inspection in nuclear power plants: Safety guide NS-G-2.6, 2002
[A2-7] IAEA safety standards series: Modifications to nuclear power plants: Safety guide NS-G-2.3, 2001
[A2-8] IAEA safety standards series: Seismic design and qualification of nuclear power plants: Safety guide NS-G-1.6, 2003
[A2-9] IAEA safety standards series: Safety classification of structures, systems and components in nuclear power plants: Safety guide SSG-30, 2014
[A2-10] IAEA safety standards series: Ageing management for nuclear power plants: Safety guide NS-G-2.12, 2009

[A2-11] IAEA safety standards series: Maintenance, surveillance and in-service inspection of nuclear power plants: Safety guide NS-G-2.6, 2009
[A2-12] IAEA Safety Standards series: Ageing management and development of a programme for long term operation of nuclear power plants: Safety guide SSG-48, 2018
[A2-13] IAEA Safety Standards series: Safety of Nuclear Fuel Cycle Facilities. Specific Safety Requirement SSR-4. 2017
[A2-14] IAEA Safety Standards series: Disposal of Radioactive Waste. Specific Safety Requirement SSR-5. 2011.

Safety reports series

A selection of relevant documents from the documents listed on the IAEA website:

[A2-15] IAEA safety reports series no. 82: Ageing management for nuclear power plants: International generic ageing lessons learned (IGALL) 2015
[A2-16] IAEA safety reports series no. 81: Development of a regulatory inspection programme for a new nuclear power plant project, 2014
[A2-17] IAEA safety reports series no. 62: Proactive management of ageing for nuclear power plants, 2009

Technical reports series

A selection of relevant documents from the documents listed on the IAEA website:

[A2-18] IAEA technical reports series: Methodology for the management of ageing of nuclear power plant components important to safety TRS 338, 1992
[A2-19] IAEA technical reports series: Guidelines for application of the master curve approach to reactor pressure vessel integrity in nuclear power plants TRS 429, 2005
[A2-20] IAEA technical reports series: Plant life management for long term operation of light water reactors - principles and guidelines TRS 448, 2006
[A2-21] IAEA technical reports series: Neutron irradiation embrittlement of reactor pressure vessel steels TRS 163, 1975

Technical documents (TECDOCs)

A selection of relevant documents from the documents listed on the IAEA website:

[A2-22] IAEA-TECDOC-189 Fracture mechanics applications: implications of detected flaws, Winterthur 3-5 December 1975, 1976
[A2-23] IAEA-TECDOC-510 Status of advanced technology and design for water cooled reactors: heavy water reactors, 1989
[A2-24] IAEA TECDOC-677 Progress in development and design aspects of advanced water cooled reactors, 1992
[A2-25] IAEA TECDOC-682 Objectives for the development of advanced nuclear plants, 1993
[A2-26] IAEA TECDOC-710 Applicability of the leak before break concept 1993
[A2-27] IAEA TECDOC-752 Status of advanced containment systems for next generation water reactors, 1994
[A2-28] IAEA-TECDOC-774 Guidance on the application of leak before break concept - Report of the IAEA extra-budgetary programme on the safety of WWER 440 Model 230 Nuclear power plants, 1994
[A2-29] IAEA-TECDOC-936 Terms for describing new, advanced nuclear power plants, 1997
[A2-30] IAEA TECDOC-968 Status of advanced light water reactor designs 1996, 1997

[A2-31] IAEA-TECDOC-977 Integral design concepts of advanced water cooled reactors - proceedings of a technical committee meeting held in Obninsk, Russian Federation, 9-12 October 1995, 1997
[A2-32] IAEA-TECDOC-981 Assessment and management of ageing of major nuclear power plant components important to safety: Steam generators, 1997
[A2-33] IAEA-TECDOC-1037 Assessment and management of ageing of major nuclear power plant components important to safety: CANDU pressure tubes, 1998
[A2-34] IAEA-TECDOC-1119 Assessment and management of ageing of major nuclear power plant components important to safety - PWR vessel internals, 1999
[A2-35] IAEA-TECDOC-1120 Assessment and management of ageing of major nuclear power plant components important to safety - PWR pressure vessels, 1999
[A2-36] IAEA TECDOC-1181 Assessment and management of ageing of major nuclear power plant components important to safety: Metal components of BWR containment systems, 2000
[A2-37] IAEA-TECDOC-1197 Assessment and management of ageing of major nuclear power plant components important to safety: CANDU reactor assemblies, 2001
[A3-38] IAEA TECDOC-1263 Application of non-destructive testing and in-service inspection to research reactors - Results of a coordinated research programme, 2001
[A2-39] IAEA-TECDOC-1303 High temperature on-line monitoring of water chemistry and corrosion control in water cooled power reactors: Report of a co-ordinated research project 1995-1999, 2002
[A2-40] IAEA-TECDOC-1341 Extreme external events in the design and assessment of Nuclear Power Plants, 2003
[A2-41] IAEA TECDOC-1347 Consideration of external events in the design of nuclear facilities other than nuclear power plants, with emphasis on earthquakes, 2003
[A2-42] IAEA-TECDOC-1361 Assessment and management of ageing of major nuclear power plant components important to safety: Primary piping in PWRs, 2003
[A2-43] IAEA-TECDOC-1390 Construction and commissioning experience of evolutionary water cooled nuclear power plants, 2004
[A2-44] IAEA-TECDOC-1391 Status of advanced light water reactor designs 1994, 2004
[A2-45] IAEA-TECDOC-1400 Improvement of in-service inspection in nuclear power plants, 2004
[A2-46] IAEA-TECDOC-1435 Application of surveillance programme results to reactor pressure vessel integrity assessment: Results of a coordinated research project 2000-2004, 2005
[A2-47] IAEA-TECDOC-1441 Effects of nickel on irradiation embrittlement of light water reactor pressure vessel steels, 2005
[A2-48] IAEA-TECDOC-1442 Guidelines for prediction of irradiation embrittlement of operating WWER-440 reactor pressure vessels: Report prepared within the framework of the coordinated research project, 2005
[A2-49] IAEA-TECDOC-1470 Assessment and management of ageing of major nuclear power plant components important to safety: BWR pressure vessels, 2005
[A2-50] IAEA-TECDOC-1471 Assessment and management of ageing of major nuclear power plant components important to safety: BWR pressure vessel internals, 2005
[A2-51] IAEA-TECDOC-1474 Natural circulation in water cooled nuclear power plants - phenomena, models and methodology for system reliability assessments, 2005
[A2-52] IAEA-TECDOC-1487 Advanced nuclear plant design options to cope with external events, 2006
[A2-53] IAEA-TECDOC-1503 Nuclear power plant life management processes: Guidelines and practices for heavy water reactors, 2006
[A2-54] IAEA-TECDOC-1556 Assessment and management of ageing of major nuclear power plant, 2007
[A2-55] IAEA-TECDOC-1557 Assessment and management of ageing of major nuclear power plant components important to safety: PWR vessel internals, 2007

[A2-56] IAEA TECDOC NP-T-3.11 Integrity of reactor pressure vessels in nuclear power plants: Assessment of irradiation embrittlement effects in reactor pressure vessel steels, 2009
[A2-57] IAEA-TECDOC-1852 Dissimilar Metal Weld Inspection, Monitoring and Repair Approaches, 2018

10. APPENDIX A3: CASE STUDIES RELATED TO INHOMOGENEITIES IN STEELS

A.3.1 At some level, heterogeneity and anisotropy are present in all the metallurgical microstructures relevant to this TAG. Normally, metallurgical processes are designed to produce products which fulfil their functions throughout their volumes and these variations are not a concern. However, the controls on the processes are not always fully effective. This appendix provides selected cases to illustrate potential problems:

- Hydrogen-induced cracking in RPV steels
- Macrosegregation in RPV steels
- Sizewell A boiler shell cracking.

Hydrogen-induced cracking in RPV steels [A3-1]

A3.2 **Plant history and 2012 outage findings**—Doel 3 and Tihange 2 are PWR nuclear power stations in Belgium which were commissioned in 1982 and 1983. The plants were built to the same design and the RPVs for both stations were produced by the same manufacturing route. In 2012, during the third ten-yearly inspection of Doel 3, thousands of defects were detected in the RPV steels and, later in 2012, inspection of the Tihange 2 RPV revealed similar defects. Power generation at Doel 3 and Tihange 2 was suspended while investigations were made into the causes of the defects and the implications for the safety of the plant.

A3.3 **Hydrogen-induced cracking**—The investigations indicated that the defects were hydrogen-induced cracks; these cracks form by the combined effects of local embrittlement and the build-up of gas pressure in inclusions on cooling. (The solubility of hydrogen in steels decreases with cooling.) The defects in the Doel 3 and Tihange 2 RPVs were associated with manganese sulfide inclusions formed from impurities in the steels and were situated in regions corresponding to residual segregation of the ingots after forging [A3-2]. Hydrogen-induced cracking:

- Can be prevented by controlling hydrogen to low levels
- Forms defects that are generally oriented parallel to the surface of a forging
- Can be found by ultrasonic inspection.

Factors for the Doel 3 and Tihange 2 RPVs that are known to promote susceptibility to hydrogen-induced cracking include:

- Hydrogen levels of 1.0 to 1.5 parts per million by weight in the ingots
- No confirmation of heat treatment to reduce the hydrogen content of the forged steels
- Removal of reduced amounts of the ingot known to be affected by positive macrosegregation with relatively high levels of alloying elements, including carbon and impurities [A3-3]
- Carbon levels in the steel (0.23% by weight) towards the upper end of the specified range (0.24% by weight max).

A3.4 ***Implications***—The reasons for the non-reporting of the defects—which were detectable and reportable by the original manufacturing inspections—have not been explained suggesting a lack of compliance with the reporting criteria. In response to the findings, WENRA recommended that its members [A3-4]:

- Review relevant manufacturing and inspection records
- Consider further non-destructive examinations of RPV steels.

The Belgian nuclear safety regulator, FANC has considered the implication of the defects, concluded that they were tolerable and allowed the licensee to continue to operate the stations [A3-5]. In response to the possibility that similar defects could affect the Sizewell B RPV, the licensee produced a safety justification for the continued operation of the plant [A3-6]. This report which made proposals for additional inspections was assessed and accepted by ONR. For Hinkley Point C, ONR has considered the metallurgical processes to be used for the primary pressure circuits and concluded that they were appropriate to minimize the risk of hydrogen induced cracking and that the inspection techniques and procedures were adequate to detect and report defects of the type reported at Doel 3 and Tihange 2 [A3-7].

Macrosegregation in the Flamanville 3 RPV head and bottom dome

A3.5 ***Regulation of pressure equipment***—The EU directive on pressure equipment (97/23/EC) excludes items specifically designed for nuclear use, failure of which may cause an emission of radioactivity, nuclear pressure equipment (NPE). The exclusion is intended to allow for the application of more stringent regulations as stipulated in France by the ESPN Order of 2005 [A3-8].

A3.6 ***Technical qualification***—The ESPN Order imposes a requirement for Technical Qualification. The objectives of Technical Qualification are to ensure that:

- Components manufactured under the conditions and manner of the qualification will respect the minimum required characteristics in their entire volume
- The manufacturing process is reproducible from one piece to another and that the results of tests performed only on the qualification component would be the same on serial components.

A3.7 ***Findings***—In 2011 ASN asked for evidence that the NPE at Flamanville 3 satisfied the requirements of the ESPN Order. In response in 2012, Areva offered proposals to qualify the Flamanville 3 RPV head and bottom dome. The proposals used destructive tests on a forging for an EPR™ RPV vessel head made by the same manufacturing route as the Flamanville 3 RPV head and bottom dome and showed (in 2014):

- Compositional variations in the forging (macrosegregation) with a region of high carbon content near the centre of the external surface. The carbon content of material near the centre of the forging (approximately 0.30% by weight) exceeded the maximum permitted by the steel specification (0.22% by weight).
- Low Charpy impact test energies. The energies for single tests were between 36 J and 64 J with an average of 52 J for material from the central region of the forging. These results are lower than the code limits (RCC-M) of 60 J for a single test and 80 J on average.

Subsequent compositional analysis of the Flamanville 3 RPV head and bottom dome showed a similar pattern of segregation to that in the forging used for the destructive tests [A3-9].

A3.8 ***Macrosegregation***—The top central region of the EPR™ RPV dome forging correspond to the last region of the ingot to solidify. Material in this region contains high levels of carbon and other segregation and should have been removed as part of the manufacturing process. While the forgings used for the EPR™ RPV head and bottom dome use an established type of steel (AFNOR 16MND5 as per RCC-M which is similar to SA 508 Grade 3), there are aspects of the forgings which will have affected the levels of segregation:

- The EPR™ ingots are relatively large: 156 000 kg ingots are used for the EPR™ RPV head and bottom dome whereas smaller ingots were used for the same components in earlier reactors.
- The steel-maker used a different casting technology for the EPR™ RPV head and bottom dome from that used previously. The EPR™ components were cast

as large ingots whereas, previously, similar components had been made using a directional solidification technique which optimized the location of segregation and the geometry of the ingot to allow the efficient removal of segregated material [A3-10].

A3.9 ***Implications***—After considering justifications for the use of the components, ASN has allowed the use of the Flamanville 3 bottom dome subject to further inspection and use of the PRV head until 2025 when it will be replaced. The problems associated with the Flamanville 3 RPV illustrate inter alia the importance of a rigorous approach to identify technical risks and mitigate them at the earliest practicable stages of major projects.

Cracking of the Sizewell A boiler shells

A3.10 ***Description of the plant***—Sizewell A Power Station (no longer operational) had two reactors, each connected to four coolant circuits. Each coolant circuit transported hot gas from a penetration in the RPV via ducts through a boiler and a gas circulator and back to the RPV. The boiler shells were cylindrical vessels with domed ends. The cylindrical regions were 18.9 m high, 6.86 m internal diameter and 57 mm thick. They were fabricated from seven courses each consisting of three plates joined by axial butt welds; the courses were joined by circumferential butt welds. The Sizewell A boiler shells were made using a Mn-Cr-Mo-V low alloy steel which offered greater strength than the carbon-manganese steels used for other Magnox primary pressure circuits. The boilers were fabricated between 1961 and 1963 and Sizewell A commissioned in 1966 [A3-11]. One of the Sizewell A boilers failed its pre-service hydrostatic test; plates salvaged from this boiler were used to construct the boiler coded 2C [A3-12].

A3.11 ***Findings of inspections and investigations***—In 1996 as part of the Periodic Safety Review, selected seam welds of the boilers at Sizewell A were non-destructively inspected. Initially, the inspections targeted the axial seam welds because there was a possibility that they were affected by reheat cracking and they were subject to higher primary stresses than the circumferential seam welds. The inspections revealed minor defects in the axial welds and further investigations revealed extensive defects in the circumferential seam welds of the pressure circuits of Reactor 2. The defects were:

- Found in three of the four boilers in the Reactor 2 pressure circuits
- Up to 25 mm deep and 4.4 m long
- Located at grain boundaries in the heat affected zones (HAZs) of the welds
- Associated with cavitation damage and oxidation; with features that indicated that the cracks formed during the stress relief heat treatments during fabrication [A3-11].

A3.12 ***Repairs to the boiler shells***—The licensee removed the cracks by a series of machining, welding and heat treatment operations [A3-13]. These were controlled to minimize deformation of untempered heat affected zones at temperatures close to 600 °C at which the creep ductility would have been low [A3-14]. The processes were assessed against the SAPs [A3-15]:

- Adaptation of a relevant nuclear code and non-compliances or deviations adequately justified (ECS.3, ECS.4, ECS.5, EMC.4, EMC.14 & EMC.30)
- Clear and auditable project safety management arrangements (SC.1 & SC.7)
- Independent monitoring, audit, review, and verification of the design, fabrication, qualification, inspection and tests and safety case (SC.1 & EMC.18)
- Redundant, diverse and qualified inspections to the highest standard, repeated at appropriate stages of the repair. Independent qualification for the inspections deemed to be most safety critical (EMC.18)
- Materials sampling and testing to address safety issues for the weld repair and return to service safety cases (EMC.1 & EMC.2 and SAPs paragraph 295)

- Optimization of repair processes to prevent cracking (EMC.5 & EMC.14)
- Avoidance or mitigation of damage to primary circuit and boiler components (EMC.5 & EMC.19).

A3.13 ***Lessons learnt*** — The inspections in 1996 showed the value of periodic inspections; the defects were not precluded by the manufacturing controls. The choice of an unusual steel for the boiler shells was a contributing factor and illustrates the importance of the use of proven materials (EMC.1 to 3); a disproportionate number of failures have occurred in Mn-Cr-Mo-V low alloy steels which require particular care in welding and fabrication [A3-15]. The licensee showed effective control of a complex repair project.

References

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11. APPENDIX A4: DEVELOPMENT OF ONR EXPECTATIONS FOR STRUCTURAL INTEGRITY OF HIGHEST RELIABILITY COMPONENTS

A4.1 This appendix outlines the development of ONR expectations for the structural integrity of highest reliability components for which specific guidance is given in §5.13 to §5.26.

A4.2 ONR's expectations have developed from the regulation of new nuclear build in the UK since the 1980s, during the Sizewell B inquiry and during the GDA process. Following the 2006 Energy review, the UK government requested ONR and the Environment Agency (EA) to assess proposed reactor designs in advance, or in parallel with, an application for a nuclear site licence. The GDA process has now been completed for the EDF/Framatome EPR™ being built at Hinkley Point C, the Westinghouse AP1000 reactor design and the Hitachi-GE UK Advanced boiling water reactor (UK ABWR). At the time of writing this TAG, the GDA of the General Nuclear System Limited UK HPR1000 was being progressed.

A4.3 The developments for Sizewell B and the GDA process are outlined below. Specific technical guidance for the GDA process has been published [A4-6], and a brief summary provided in this Appendix.

Sizewell B

A4.4 Sizewell B was the first civil PWR constructed in the UK. A succession of groups reviewed the structural integrity of the plant and made recommendations on its manufacture, design, inspection, material monitoring and quality assurance [A4-1, A4-2, A4-3 & A4-4]. These are captured in the SAPs on the highest reliability SSCs, EMC.1 to EMC.3.

A4.5 The structural integrity safety case for Sizewell B set a precedent for the inference of high levels of structural integrity; it comprised [A4-5 & A4-6]:

- 'Achievement of integrity' based on compliance with an established design, construction and inspection code (supplemented with additional measures in design, quality assurance, materials and inspection);
- 'Demonstration of integrity' with the emphasis on showing defect tolerance with the support of qualified inspections.

The requirements are achieved as follows.

Design codes

A4.6 The strengths of design codes are generally well understood (design by analysis, degradation, brittle fracture safe operation, comprehensive QA, inspections, etc.). However, their weaknesses are less well understood and may not always represent what is safe so far as reasonably practicable. As such, ONR views a design code as a minimum requirement for a highest reliability SSC for which additional measures may be applied according to the severity of the consequences of failure. For example, in addition to meeting the design code requirements, the Sizewell B design includes provisions for effective inspection; see §5.91.

Fracture mechanics assessment

A4.7 The expectation is that a safety case where a highest reliability claim is made, the design code assessment is supported by a defect tolerance assessment which uses fracture toughness data for the specific steels and weld metals of concern. In general, the expectation is that lower bound fracture toughness data are used for the assessment. In some situations, an allowance for stable tearing may be appropriate for low frequency faults subject to safety case justification.

Manufacture

A4.8 In addition to meeting the requirements for manufacture and testing (§5.58 to §5.68), there may be scope for ALARP improvements to reflect more recent metallurgical practice, for example, by controlling levels of alloying and tramp elements within the permitted ranges [A4-7].

A4.9 A licensee must be able to demonstrate effective control of welding operations. Systems must be in place to qualify welding processes in advance and ensure that the requirements are met in practice. Welders must be appropriately trained and welding consumables rigorously controlled.

Inspection

A4.10 There is an expectation for the manufacturing, pre- and in-service examination and testing inspection to be sufficiently redundant, diverse and qualified. Appendix A5 details ONR's expectations for inspection qualification.

A4.11 Design code based radiographic examination may not be sufficient. Objective based NDT may necessitate the use of ultrasonic techniques at the time of manufacture and during service.

GDA process

A4.12 A set of expectations has been developed for structural integrity within the GDA process [A4-6] within which key elements are:

- The safety classification process and its linkage to the assignment of appropriate design, construction and inspection codes — In addition, if highest reliability is claimed in the safety case, ONR expects additional measures beyond normal practice i.e. beyond the provisions of established nuclear design, construction and inspection codes.
- A defect tolerance assessment for a sample of the limiting locations in highest reliability SSCs — This is a conservative assessment using lower bound material properties with plans for fully representative fracture toughness testing. For these assessments, an acceptable target is that the defect size margin (DSM) defined as the ratio of the defect size that can be tolerated at the end of life to the detectable defect size with allowances for degradation in service by fatigue and by embrittlement is at least two; see §5.91 to §5.96.and [A4-8].
- Reduction SFAIRP of the number and length of welds (EMC.9 & EMC.10). The extent to which this is appropriate is limited by the practicalities of producing large forgings with acceptable properties; see §5.63 and Appendix A3. The requesting party should strike an appropriate balance between eliminating or reducing weld volumes and achieving adequate material properties in large thick section forgings.

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12. APPENDIX A5: INSPECTION QUALIFICATION IN RELATION TO NDT

A5.1 Non-destructive testing (NDT) is an important tool for demonstrating the structural integrity of SSCs (hereafter simply referred to as components) that are important for nuclear safety. Accordingly, ONR looks to nuclear licensees to have suitable arrangements in place for ensuring that NDT is clearly specified and for providing assurance in the capability and reliable implementation of NDT. These arrangements need to be commensurate with the role of the NDT and the structural integrity classification of the SSC.

A5.2 This appendix describes ONR expectations for demonstrating the reliability of NDT through a process of inspection qualification. In particular, it concentrates on those instances where high reliability of NDT is required to support a structural integrity safety case.

A5.3 The appendix describes:

- the relationship between the structural integrity safety case and NDT reliability;
- the objectives of qualifying NDT
- the underlying principles for qualifying NDT
- the process of inspection qualification that ONR considers as relevant good practice for UK nuclear reactors.

Relationship to SAPs

A5.4 The SAPs that are relevant to NDT and specifically inspection qualification are those on the integrity of metallic SSCs listed in §4.1 of the main document (EMC.1 to EMC.16 & EMC.18 to EMC.34) and:

- EAD.2 (Lifetime margins) - Adequate margins should exist throughout the life of a facility to allow for the effects of materials ageing and degradation processes on structures, systems and components
- ECS.3 (Codes and standards) - 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 codes and standards
- EMT.1 (Identification of requirements) - Safety requirements for in-service testing, inspection and other maintenance procedures and frequencies should be identified in the safety case
- EMT.2 (Frequency) - Structures, systems and components should receive regular and systematic examination, inspection, maintenance and testing as defined in the safety case
- EMT.3 (Procedures) - Commissioning and in-service inspection and test procedures should be adopted that ensure initial and continuing quality and reliability
- EMT.6 (Reliability claims) - Provision should be made for testing, maintaining, monitoring and inspecting structures, systems and components (including portable equipment) in service or at intervals throughout their life, commensurate with the reliability required of each item.

Relationship between the structural integrity safety case and NDT reliability

A5.5 NDT plays an essential role in demonstrating the structural integrity of SSCs that are important for nuclear safety. Accordingly, ONR looks to nuclear licensees to have suitable arrangements in place for ensuring that NDT is clearly specified and that the requirements for providing assurance in the capability and reliable implementation of NDT are defined.

A5.6 ONR expects the structural integrity safety case to define:

- The objective of the NDT in relation to structural integrity
- The performance objectives and reliability required of the NDT
- The measures taken to provide assurance that the NDT is capable of delivering the defined objectives under site conditions.

Objectives of NDT

A5.7 NDT can be used for a range of purposes and it is important that the NDT method is selected accordingly. Generally, the structural integrity safety case will define the overall objectives of the NDT, along with details of the defects/degradation that is of interest. Examples of NDT objectives include:

- Detection of unacceptable manufacturing defects
- Detection and evaluation of defects during in-service inspection
- Establish whether existing defects are growing
- Distinguishing between volumetric and planar defects.

A5.8 In cases other than where high reliability is required, it may be sufficient to refer to a code or suitable standard to provide this information.

Performance objectives & reliability

A5.9 In addition to defining the overall objectives of the NDT, the safety case is expected to define the performance objectives, such as the detection and rejection of defects above a specified size. Here, it may be sufficient to refer to codes and standards for acceptance criteria.

A5.10 The safety case should define the safety significance of the SSC and the role that NDT plays in assuring its structural integrity. A specific claim of high reliability NDT is expected where the NDT plays an important role in supporting the structural integrity of a ‘highest reliability’ SSC (SAPs EMC.1 to EMC.3). Here, the term high reliability NDT means that the NDT has an inherent capability demonstrated through inspection qualification along with specific measures to ensure the NDT is applied as intended. This guide is focused on the aspect of inspection qualification with some discussion on those additional measures that are taken to assure the reliable application of the NDT.

The objectives of qualifying NDT

A5.11 In many cases, NDT may be considered as being ‘code-based’. This means that the NDT requirements, method and acceptance criteria are specified by a suitable code or standard. Where high reliability is claimed for the NDT, it is helpful to use the term objective-based NDT as the objectives for the NDT are defined by the safety case and refer to a specific SSC. These are usually presented in the form of an inspection specification.

A5.12 Inspection qualification aims to provide an appropriate level of confidence that the NDT system is capable of delivering the specified objectives under site conditions. Here, the term NDT system is the combination of the NDT procedure (techniques), equipment and personnel.

Criteria for high reliability NDT

A5.13 NDT for which a high reliability claim is made should meet several criteria, including:

- A clearly defined inspection specification is produced that describes all of the relevant parameters for the NDT and includes items such as defect descriptions, acceptance criteria and overall performance objectives.
- The NDT method and techniques are based upon well-established physical principles. An example here is the use of specular reflection from the face of a defect for ultrasonic inspection.
- The NDT system is capable of achieving good margins for detection and evaluation.
- The range of parameters for which the procedure is valid, is understood and can be specified and controlled.
- A clear and complete NDT procedure describes all of the necessary instructions. Here, a test of the adequacy of the procedure is whether several NDT operators would apply the inspection in the same way and would produce broadly the same results.
- The NDT personnel are highly proficient in performing their allocated roles. This is achieved through, specific training and a rigorous demonstration of competence.
- The NDT system provides high quality records that provide assurance that the NDT has been conducted in line with the procedure.

The process of inspection qualification

A5.14 The purpose of inspection qualification is to provide an appropriate level of confidence that the NDT system is capable, under site conditions, of meeting the pre-defined inspection objectives, usually provided in an inspection specification. It is important that the inspection objectives are driven by the requirements of the safety case.

A5.15 There are few available schemes for assuring high reliability of NDT and of these, it is only the ENIQ (European network for inspection and qualification) methodology [A5-1] that is judged to be well suited to objective based NDT and that it is aligned with the underlying principles for high reliability NDT described above. It is worth noting that the ENIQ Methodology evolved from the process, known as ‘inspection validation’, which was used to provide assurance in the manufacturing and in-service inspection of incredibility of failure components of Sizewell B.

A5.16 As a framework document, the ENIQ Methodology provides general principles that:

- Can be adopted to meet the specific needs of countries
- Can be applied to NDT performed in manufacture and in-service
- Can, in principle, be applied to any NDT method.

A5.17 The methodology document is accompanied by a series of recommended practices that provide guidance on the major qualification items and activities. While it is accepted that the qualification process can be tailored according to the specific application, there are some general features of the ENIQ Methodology that are expected to be applied and are given below:

- An inspection specification defines all of the parameters that are relevant to the NDT, including the defect details, component design and access and the performance requirements.
- The qualification of the NDT procedure and the qualification of personnel are usually separated.
- The qualification of the NDT procedure is a combination of a written assessment in a technical justification and practical trials.

- NDT personnel are usually qualified through blind trials and often a combination of blind trials and other assessments.

A5.18 A qualification body is established to assess the NDT procedure and the technical justification, conduct practical trials and assess the NDT personnel. While variable models exist for the qualification body, it is essential that it is able to exercise its judgment free from external pressures; an independent third party body is preferred.

A5.19 An important part of the technical justification produced by the NDT designer is to define the physical principles for the inspection and to set down the basis for the design and in this respect, this ‘physical reasoning’ is part of the inspection design process. From a regulatory standpoint it is important that the NDT is based upon sound physical principles (for example specular reflection from the face of a defect for ultrasonic inspections) and that these are clearly understood. This understanding will enable the NDT designer to identify the ‘essential parameters’ and to ensure that they are adequately controlled in the NDT procedure.

A5.20 Personnel certification according to ISO 9712 demonstrates a certain level of competence but this is judged not to be sufficient for high reliability nuclear inspections. Where highest reliability is sought then NDT personnel are qualified by applying the ENIQ Methodology with qualifications usually being against a specific NDT procedure (or group of procedures). Under the ENIQ Methodology, personnel qualification is undertaken through a strong element of blind trials and is conducted by the qualification body.

A5.21 Inspection qualification is an important process, but it is only able to provide confidence that the NDT system (procedure, equipment and personnel) is capable of delivering the required inspection objectives. Rigorous quality assurance, accompanied by surveillance of the NDT, is essential to ensure that this potential is realised in practice.

References

[A.5-1] European network for inspection qualification (ENIQ), 'European methodology for qualification of non-destructive testing 3rd issue,' ENIQ Report 31, EUR 22906, 2007 (CM9 2019/365173)