ONR GUIDE			
INTEGRITY OF METAL STRUCTURES, SYSTEMS AND COMPONENTS			
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INTRODUCTION

1.1. ONR has established its Safety Assessment Principles (SAPs) which apply to the assessment by ONR specialist inspectors of safety cases for nuclear facilities that may be operated by potential licensees, existing licensees, or other duty-holders. The principles presented in the SAPs are supported by a suite of guides to assist ONR's inspectors in their technical assessment work to support the making of regulatory judgments and decisions. This technical assessment guide (TAG) is one of these guides.

1.2. The TAG (Revision 5) is revised and updated; in particular, to give:

- Advice on:
  - The international response to the Fukushima nuclear accident (Appendix A1).
  - Inhomogeneities in steels and technical qualification in (§5.29 & §5.30) and (Appendix A3)
- Updated guidance on:
  - The purpose of assessment and safety categorization and classification (§5.3)
  - The manufacture of highest reliability components and structures (§5.56 to §5.64)
  - Lifetime records (§5.70)
  - Operational experience (§5.100 to §5.103).

2. PURPOSE AND SCOPE

2.1. This TAG provides ONR inspectors with additional guidance and interpretation of SAPs EMC.1 to EMC.34 which are concerned with the integrity of metal structures, systems and components (SSCs). These SAPs cover process-containing SSCs such as pressure vessels, storage tanks, pipes and valves. The scope of the TAG includes new and existing nuclear licenced facilities through all phases of their operation; it excludes metallic structures within the civil engineering ambit such as building frames, pipe bridges and crane supports.

2.2. Safety cases for nuclear installations are often complex and can require assessment by a range of specialisms to judge whether they are adequate and ALARP. Therefore the assessment of the structural integrity aspects of a safety case is rarely undertaken in isolation or without reference to other relevant SAPs and TAGs. Other TAGs that inspectors may wish to refer to are listed in §4.3 and §4.4.

2.3. 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 and A2 respectively.

2.4. Appendix A3 provides cases studies that illustrate the significance of manufacturing processes to the integrity of metal structures, systems and components (SSCs).

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 authorized and other suitably qualified and experienced persons)
- LC 13 (Nuclear safety committee)
- LC 36 (Organisational capability)

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

SAPs

4.1. 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 [1]—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)

4.2. The sequence of the SAPs on the integrity of metal SSCs in [1] (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 dropped.
- SAPs EMC.7 to EMC.16 and EMC.18 to EMC.34 are given in the sequence of processes applied to a component.
- A decision has been made to give advice on manufacturing inspection (in EMC.27 to EMC.30) after all guidance on manufacture and installation. (Rather than the location of the discontinued SAP EMC.17.)

- Editorial policy for the SAPs is to retain numbers for SAPs that are largely unchanged.

4.3. 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 (paragraphs 519 to 538). Where the SSC forms part of a core support structure, SAPs ERC.1 to ERC.4 (paragraphs 539 to 557) should be considered.

4.4. 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.17 (External and internal hazards)
- EKP.1 (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 and 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.7 (Safety case processes)

WENRA reference levels and IAEA safety standards

4.5. Appendix A1 of this TAG considers the WENRA reference levels for reactor safety; decommissioning safety and waste and spent fuel storage safety. Appendix A2 considers IAEA documents: from the Safety Standards Series; Safety Reports Series; Technical Reports Series and Technical Documents Series.

5. ADVICE TO INSPECTORS

Introduction

5.1. This guide provides advice on the assessment of the structural integrity aspects of safety cases. The relevant SAPs (§4.1) 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 where gross failures can be discounted.

5.2. Guidance on what ONR considers to be an effective safety case is given in the regulatory assessment of safety cases section of the SAPs (SC.1 to SC.8) and the supporting TAG:

- NS-TAST-GD-051 The purpose, scope and content of nuclear safety cases.

In addition, the inspector may find the following TAGs relevant to structural integrity assessments:

- NS-TAST-GD-005 Guidance on the demonstration of ALARP
- NS-TAST-GD-030 Probabilistic safety analysis
- NS-TAST-GD-033 Licensee management of records

- NS-TAST-GD-049 Licensee use of contractors and intelligent customer capability
- NS-TAST-GD-077 Supply chain management arrangements for the procurement of nuclear safety
- NS-TAST-GD-094 Categorization of safety functions and classification of structures systems and components.

5.3. The purpose of 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. The inspector needs to begin by understanding the nuclear safety category of the SSC (ECS.1); this will determine the safety classification (ECS.2) and the appropriate codes and standards (ECS.3). 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 functions contributing to nuclear safety.'*

On safety functions 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 §5.7.4.

5.4. The highest demands are placed on the 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). Assessment of this type of safety case is dealt with in §5.17 to §5.22. This type of structural integrity safety case is discussed in the context of Sizewell B in [2 & 3] and the German Basis Safety Concept (BSC) or Break Preclusion Concept in [4]. Not all structural integrity safety cases need to claim the discounting of gross failure; cases that can be made with less demand on structural integrity do not carry the same burden of requirements.

5.5. The safety function categorisation and safety classification of SSCs, as outlined in §5.3, determines 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 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. This shading of requirement is fundamental to the 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. SAP NT.1 Targets 8 and 9 give upper limits on risk of exposure to ionizing 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 relevant. 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.

5.7. 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) 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 [5]. 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.

5.8. Structural integrity encompasses a number of technical areas including metallurgy and materials property testing, welding engineering, stress analysis, fracture mechanics and examination techniques. An inspector may be experienced in a number of these. Nevertheless, the inspector should be alert to those aspects of an assessment where they may need to consult with colleagues; inspectors should avoid giving undue attention to those aspects with which they are most familiar. It is unlikely that a structural integrity safety case could be made on one technical specialty. At the same time, some structural integrity safety cases present complex technically challenging arguments and evidence and the inspector may reach a conclusion on the acceptability of the case using a different weighting of the features of the safety case to that presented by the licensee.

5.9. 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.'*

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.

Structural integrity philosophy

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 (§5.3), 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, inclusion in a probabilistic safety analysis (PSA) might be indicative or nominal and rudimentary compared with other aspects of the PSA.

5.12. For structures and components classified as highest reliability, the inspector should invoke principles EMC.1, EMC.2 and EMC.3 to reflect the radiological consequences of failure. EMC. 1 addresses two particularly important aspects:

- The SSC should be as defect free as possible
- It should be demonstrated to be defect tolerant.

In particular, the critical / limiting crack sizes need to be larger than the defect size that can be reliably detected and characterized by the applied examination techniques (how much larger will depend on the overall case and is an important aspect for judgment by the inspector). This wording which is in terms of crack-like defects can be generalized to other forms of degradation; see SAPs paragraph 283 for the definition of the term ‘defect’ as used here. And 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 SAPs paragraph 286; the SAPs accommodate the likely necessity to assess this sort of safety case in some circumstances.

5.13. EMC.2 calls for the assessment to include a comprehensive examination of relevant scientific and technical issues and to take account of available precedent.

5.14. To achieve these fundamental requirements, several related and independent arguments should be used, based on the following (see EMC.3 - many of the following contribute to the defect tolerance of SSCs or the management of aspects that affect defect tolerance):

(a) The use of sound design concepts and proven design features

(b) A detailed design loading specification covering normal operation, plant transients, faults, 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) Application of high standards of manufacture, including manufacturing inspection and examination

(g) High standards of quality assurance throughout all stages of design, procurement, manufacture, installation and operation

(h) Pre-service and in-service examination to detect and characterize defects at a stage before they could develop to cause gross failure

(i) Defined limits of operation to ensure the facility is operated within the limits of the safety case. Where appropriate, limits of operation should be supported by protection systems, for instance overpressure protection

(j) In-service monitoring of facility operational parameters

(k) In-service materials monitoring schemes

(l) A process for review of facility operation to ensure the facility is operated and materials performance is within the assumptions of the safety case

(m) A process for review of and response to deviations

(n) 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

(o) 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.

5.15. 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 [2] and [6].

5.16. For SSCs that are not of major safety significance, the above list of requirements is also relevant, though the stringency of their application should reflect the lower safety categorisation of the item (SAPs paragraphs 297 to 300). Principles covering those various requirements are presented below.

Highest reliability SSCs: Discounting gross failure

5.17. 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 mitigating the effects of the structural 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.

5.18. 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 emphasized:

> *'…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…'*

These are indicative failure frequencies. A safety case that discounts gross failure cannot be a 'formal proof' of such reliability levels. See §5.100 to §5.103 for a review of 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 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 and 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. 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).

5.22. As background to the assessment of structural integrity safety cases that discount gross failure, the inspector may wish to consider the Technical Advisory Group on Structural Integrity (TAGSI) response to an ONR question on the matter [7].

Safety function and safety categorisation

5.23. The safety functional requirements of SSCs should be identified from the fault schedule; see SAPs paragraphs 407 and 643, and the appropriate safety categorisation 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, and on the requirement to meet the functional requirement for the proposed life of the facility. From this, the appropriate standards of design, manufacture, installation and testing, in-service maintenance, inspection and testing, and operation can be derived. 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.

5.24. ECS.2 requires that SSCs should be categorized based on the consequences of failure and of the failure frequency requirements of the safety case. ECS.3 then suggests appropriate safety case requirements, and thus assessment requirements.

5.25. The 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 protective devices that can 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 a principal safety feature.

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

5.26. 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. Guidance on safety classification, codes and standards is provided in ECS.1 to ECS.5, paragraphs 158 to 179, paragraphs 172 to 173, paragraph 191 and EAD.2. The requirements depend on the safety categorisation of the structure. Design activities should be subject to procedural control (EAD.4).

5.27. All loadings for operation, credible faults and tests should be identified and the magnitudes specified (EMC.7, EMC.11). Load combinations should be defined. SAPs EHA.1, EHA.3 to EHA.5 and EHA.7 cover external and internal hazard loads. 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.

5.28. For pressure boundary and other load bearing structures, the use of appropriate British Standards, or International Standards might well 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. Most design codes express limits in term of stress. However, for non-pressure boundary structures *e.g.* core support structures, functional limits on displacement may be important.

5.29. 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 and 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 the case study on the Flamanville 3 domes in Appendix A3. The inspector should balance the minimization of welds against the potential difficulties of producing large forgings with acceptable properties throughout their volumes (EMC.5).

5.30. The design should be supported by stress analyses and, if necessary, model tests, to validate the methods used, to demonstrate that adequate margins against failure are maintained throughout the plant life (EMC.32). As appropriate, prototype testing of unusual design features such as secondary retention devices and technical qualification of any novel metallurgical processes should form part of the structural integrity safety case. Consideration should be given to the applicability of any tests to plant conditions and uncertainties, particularly those associated with environmental effects when reliance is placed on testing with simulated conditions. Analyses and tests need to be done under a quality process that will provide a basis for relying on the results. The adequacy of margins needs to be considered in the light of the perceived accuracy, reliability and conservatism of analysis and test results.

5.31. 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.

5.32. The design should take due account of degradation processes, including irradiation embrittlement, corrosion, erosion, creep, fatigue and ageing, and for the effects of the chemical and physical environment. The potential for interaction effects 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. Of particular importance are degradation mechanisms in SSCs that are difficult or impractical to inspect in service. In these cases, conservative estimates should be included in the design and appropriate surveillance schemes specified. 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.

5.33. 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. 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).

5.34. For existing plant, it is recognized 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, or be demonstrated that any shortcomings are not significant in terms of 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 assessment of 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 needs to be carefully considered.

5.35. Safety submissions for existing plant 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.

Load analysis - The analysis of all conditions within the design basis

5.36. The safety case should include an analysis of the potential failure modes for all conditions arising from design basis loads. It may also be appropriate to consider the resilience of structures or components to beyond design basis events.

5.37. The objective of the analysis is to demonstrate that the structure is 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 an adequate 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 and the possibility of disruptive failure without warning should be remote (EDR.1, EMC.7 & EMC.11).

5.38. For infrequent events the inspector may need to consider whether there is scope for alleviation of the most rigorous requirements for pessimism in the structural integrity analysis. In such cases, the safety case should provide a suitable justification for any relaxation. In terms of stress analysis, it may be reasonable to use stress limits that increase as the likelihood of the loading decreases. That is lower stress limits would apply to normal operating conditions, and higher stress limits to infrequent fault loading conditions.

5.39. For fracture mechanics analyses it may be appropriate for normal operating and frequent fault conditions to base margins on initiation fracture toughness. However for infrequent fault conditions, it may be appropriate to base margins for results of fracture mechanics analyses on a level of fracture toughness enhanced by limited stable tearing. If such assessments are used, they can usefully be supported by a sensitivity study showing margins based on initiation fracture toughness. Values of fracture toughness enhanced by limited stable tearing must be supported by valid fracture toughness test data up to at least the extent of tearing invoked in the safety case (EMC.34, paragraph 318).

5.40. The complexity of the analysis will be dependent on the safety categorisation of the SSC. For the highest category, this might include finite element stress analysis and, where failure by crack growth is concerned, a fracture mechanics assessment in accordance with recognized procedures such as R5 [8], R6 [9] or BS 7910 [10]. It is important that the inspector ensures that analysis codes and procedures are adequately verified and validated for the particular application as required by EMC.32, EMC.34 and SAPs paragraph 316. For lower category structures, compliance with appropriate national and international standards may be sufficient.

5.41. The purpose of the analysis should be to demonstrate that the structure is tolerant to any actual or postulated degradation or defects that may remain after manufacture or that may develop during service. In the case of crack-like defects, the submission should show that there is an adequate margin between the size of defects capable of being detected by the examination techniques used and the critical defect size. The size of an 'adequate margin' should be judged on a case-by-case basis depending on the overall safety case for the particular SSC.

5.42. The inspector should establish that data used in analysis is demonstrably 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 based on R5 [8], R6 [9] or BS 7910 [10] *etc*. is concerned.

5.43. This TAG does not recommend minimum acceptable reserve factors (*e.g.* for R6 analysis results), since the values of margins are dependent on the conservatisms in the input data. A low margin is not likely to be acceptable without substantial justification. Sensitivity studies to establish the effects of variations in the assessment parameters assist the assessment of the safety case. Generally, safety cases for components where the dominant failure mechanism is due to crack-like defects should not rely entirely on a fracture mechanics analysis. This should be only one element of the case that could include consideration of conservative design, the use of known materials, original manufacturing quality and testing, metallurgical investigations and examinations demonstrating negligible or at least acceptable defect growth, and analysis to demonstrate defect tolerance at the end of life.

5.44. The inspector should ensure that margins against failure are apparent and adequate in the context of the overall safety case. The inspector should assess the various factors that affect these margins (*e.g.* stress, materials properties and inspection findings) to reach a conclusion on the adequacy of a safety case taking due note of previous assessment findings.

5.45. SAPs EMC.12 and EMC.23 require that the operating regime ensures that metal pressure boundaries exhibit ductile behaviour when significantly stressed. In addition to this, the ONR statement [11] on operation of reactor pressure vessels made of ferritic steel provides a formal view on the need for vessels to be operated on the upper shelf of fracture toughness. This states that ferritic steel RPVs must, for normal steady-state operation, operate on the upper shelf. 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.

5.46. 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 structures, 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.

5.47. Proof pressure test arguments might be used to show that defects which could have survived the 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 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 emphasized 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 was 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.

5.48. 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 and EMC.11). Load combinations should be defined. EHA.1, EHA.3 to EHA.5 and 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. Failure of structures may give rise to hazards such as missiles, steam or hot gas release, collisions, pipe-whip, *etc*., 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 hazards on safety related structures, and of the secondary effects of structural failure.

5.49. The hazards posed by earthquakes can present some difficulties particularly for existing structures. Earthquake loading can be included in the design specification for new plant and analysed in the design substantiation. Existing structures may have been designed and constructed prior to seismic qualification being required, or may have been qualified to a less rigorous standard than that required for new structures.

5.50. 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 §5.17 to §5.22.)

5.51. A safety case should show there is no disproportionate increase in risk for an appropriate range of events that are more severe than the design basis event, EHA.7.This implies that SSCs where the claim is gross failure can be discounted, need to be shown to be capable of withstanding the loads associated with events whose frequency of occurrence does not exceed 1 in 10 000 years (at least a robust case for ‘no cliff-edge’ effect), unless it can be shown that the frequency of an event is demonstrably below once in 10 million years, EHA.1 with paragraph 235.

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

Materials: The use of proven materials

5.53. It is important to 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.

5.54. The inspector should consider seeking confirmation that all metallurgical processes (including: casting, degassing, ladle processing, forging, 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 and EMT.6) across the full range of environmental conditions expected throughout the identified lifetime of the plant.

5.55. 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 likely material performance given the modified understanding, and establishes the implications for the performance of the structure. This may involve additional examinations, material sampling and testing, metallographic examination, testing of archive material, and simulation of material behaviour in order to improve confidence in the future 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 adequacy of data derived from surveillance samples should be examined, as appropriate, to gain assurance that the data accurately represents the component damage state, recognizing the inherent scatter in determination of most materials properties. The inspector should examine the safety case for these aspects and look for commitments for examination and monitoring covering expected and unexpected phenomena (sometimes referred to as 'safety case' based and 'speculative' examinations or monitoring).

Manufacture, inspection and testing

5.56. 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

5.57. The material specification, manufacturing processes and inspections should be suitable and should ensure that the SSC is free from significant defects and tolerant of any remaining defects (ECS.3 with paragraph 169 and EMC.5, EMC.6). 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 and EMC.9).

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

(a) The manufacturing processes and inspections are carried out in accordance with approved procedures, *e.g.* using an approved weld procedure and welders qualified for that procedure, weld repair procedures, inspection procedures and controls and the residual element content and homogeneity of steels.

(b) Appropriate third party inspection of manufacture and examination is specified to ensure that a high standard of workmanship has been achieved (EMC.14 and EMC.18). Examinations of welds in highest reliability SSCs should be redundant, diverse and qualified. Pre-service inspections should be carried out at a late stage in the period prior to operation, when the plant is in a state essentially as for normal operation.

5.59. The starting point for metallic SSCs important to nuclear safety is compliance with the relevant design codes and specifications. For highest reliability SSCs for which the likelihood of gross failure is discounted, see §5.17 to §5.22, there is an expectation that further 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.

5.60. 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.

5.61. For existing plant:

- 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 recognized 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.
- When considering modifications, new components 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).

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 §5.29 and 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 the examination of the quality of manufacture should include a review of manufacturing concessions for deviations from the original specification.

Quality assurance (QA)

5.65. There should be appropriate quality assurance throughout all stages of design, procurement, manufacture, installation, commissioning, operation and decommissioning; see EQU.1. Quality assurance arrangements are also required for production of the safety case.

5.66. The licensee should use, and require its contractors to use, formal QA procedures to specify the quality and organizational 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 recognized 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, operation and maintenance activities.

5.67. From experience of where issues can arise, the inspector may wish to check the licensee-contractor interface and other organization-to-organization 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.

5.68. 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.

5.69. 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; for example see the Doel 3 Tihange 2 case study in Appendix 3.

5.70. For steels used in the highest reliability SSCs, lifetime records should be comprehensive, indicating *inter alia*: the identities of the heat and quality plan, elemental analysis data for the ladle and the product, relevant mechanical test data, the results of non-destructive examinations and the metallurgical processes including ladle treatment, degassing, casting, forging and heat treatment (EMC.20).

5.71. 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; (MS.2, MS.4, EHF.8 and SAPs 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).

Inspection: Pre-service and in-service examination and in-service monitoring

5.72. 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 aging 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.

Manufacture and installation should be subject to appropriate third-party independent inspection to confirm that processes and procedures are being followed (EMC.18).

5.73. In general, inspection requirements should be identified in the safety case and be incorporated into the Maintenance Schedule if appropriate.

5.74. Inspection provides an important element in establishing the integrity of SSCs that are required to have the highest reliability (gross failure so unlikely it can be discounted). 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 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).

5.75. In-service examinations should be carried out where they are reasonably practicable to enable the present condition of the structure 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 high integrity SSCs the examination procedures 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.29 and EMC.30). 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 suitably qualified and experienced personnel. 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 previous operating experience (see §5.100 to §5.103) 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. Examination shortfalls should be clearly identified. For example, it may not be possible to examine 100% of a weld because of access difficulties. The implications of the inability to examine areas of welds should be addressed in the assessment of the significance of any defects found or defects that could exist in the unexaminable areas.

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 and 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.

Materials monitoring - the provision of in-service materials monitoring

5.83. 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. Significant extrapolation of data should be avoided. Extrapolation might be in time or to similar base and weld materials. 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.

5.84. 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.

Leak detection and leak-before-break

5.85. Where high reliability in structural integrity needs to be claimed and justified, a 'leak-before-break' argument may not be appropriate as the main thrust of the safety case argument. However, it depends on what is in the argument, rather than simply the label attached to it. For very high integrity (for instance where there is no 'line of protection' for the consequences of failure), a 'No Break' argument or a 'No Leaks or Break' argument might best summarize or label the sort of structural integrity safety case required. If some consequences are still protected, for example, loss of fluid by providing emergency injection, but other consequences are not, for example pipe whip and jet forces, the inspector should expect the licensee to present a clear justification for this approach.

5.86. From operational experience (see §5.100 to §5.103), incidents of sub-critical crack growth in boiling water reactor (BWR) and pressurized water reactor (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 [12 & 13]. However, this operational experience does not amount to a safety case.

5.87. Leak detection arguments and leak-before-break arguments might be provided to support pressure boundary structural integrity safety cases. Originally, the term leak-before-break referred to the situation where a defect has been sized, for example by examination, and it was argued that such a defect—were it to grow—would lead to a leak rather than a break. Its usage has widened to include leak detectability, the argument that for a range of through-wall defects, leaks can be detected while the defect remains stable.

5.88. 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.* [14]) 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.

5.89. 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. See §5.39 regarding loading condition frequency and corresponding measures of fracture toughness. Factors that may need to be considered include the potential for debris blockage of the 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.

5.90. Leak-before-break 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 A3).

5.91. Obviously, leak detection capability is fundamental (EMC.25, EMC.26 and paragraph 312). The safety case should explain the leak detection system and identify the sensitivity, reliability, 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. 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.

5.92. 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.

Ageing and degradation

5.93. 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 recognized and routinely considered in structural integrity evaluations; see §5.30, §5.32, §5.34, §5.47, §5.54, §5.55 & §5.60. 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.

5.94. 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.

5.95. The safety case for nuclear facility SSCs needs to include a suitably conservative consideration of the effects of ageing and degradation on safety margins throughout plant life, including decommissioning.

5.96. At the design stage, potential ageing and degradation mechanisms should be identified and stated as part of the design specification. With the mechanisms defined, a conservative estimate should be made of the minimum safe working life of the SSCs. 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).

5.97. It is to be expected there will be uncertainties in material properties and plant parameters required in the estimation of a safe working life. 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. Periodic review during service should use evolving in-service information to update the predicted minimum safe working life (EAD.2, EAD.3 and EAD.4).

5.98. 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. Over the course of a number of years, new approaches may be adopted to assist in supporting claims of structural integrity. An example over the last few decades has been the increasing use of fracture mechanics to demonstrate the resistance to crack growth in metallic SSCs. Current methods (*e.g.* fracture mechanics procedures) can be applied to existing components and structures. One practical limitation can be whether appropriate materials data is available for an existing component or structure to permit the plausible use of advance analysis techniques (EAD.5).

5.99. 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.

Operational experience data

5.100. 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 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.

5.101. 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.* [14 & 15]. 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.

5.102. 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?

5.103. 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

5.104. 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.

5.105. 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.

General advice

5.106. The inspector should consider, as appropriate for the key safety issues, the elements set out in paragraphs 280 to 300 of the SAPs to the appropriate depth to establish whether the design, load analysis, materials, standards of manufacture, inspection and testing, quality assurance standards, protection systems, and provisions for material monitoring, maintenance and inspection provide the necessary confidence that the safety functional requirements will be met.

5.107. It should be emphasised that the adequacy of the integrity of metal SSCs relies to an extent on each of the factors outlined, and the inspector should apply engineering judgment to the overall safety case before coming to a view on its acceptability. Structural integrity safety cases tend to be multi-legged and each leg of the argument needs to be considered before coming to a view on the overall adequacy of any case. Due consideration should be given to the potential for common mode failure mechanisms and factors that affect more than one leg of a multi-legged argument.

5.108. The inspector is not expected to repeat the analysis provided by the licensee, though sample checks may be appropriate. The assessment overall will be a sampling process. The inspector may wish to examine the licensee's process for developing the safety case to gain confidence in the content and claims of the safety case. 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 and 457 of the SAPs).

5.109. This process relies on engineering judgment. 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 [1]. 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 make alternative provisions to ensure safety.

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6. REFERENCES

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

[2] *Geraghty J E, 'Structural integrity of Sizewell B - The way forward,' Nuclear Energy, Vol. 35, No. 2, pp. 97-103, 1996*

[3] *Bullough R., et al, 'The demonstration of incredibility of failure in structural integrity safety cases,' International journal of pressure vessels and piping, Vol. 78, No. 8, pp. 539-552, 2001*

[4] *Roos E, Herter K-H, Otremba F, Metzner K-J, General concept of the integrity of pressurized components,' Trans. SMiRT 16, Division O, Paper 1725, (www.iasmirt.org/smirt/16/transactions) 2001*

[5] *ONR, Enforcement policy statement - ONR-ENF-POL-001 Revision 0, 2014 (TRIM 2014/127037)*

[6] *Sizewell B Reactor Pressure Vessel, Special Issue of Nuclear Energy, Vol. 31, No. 6, pp. 409-453, 1992*

[7] *Technical advisory group on structural integrity (TAGSI), TAGSI response to NII questions on incredibility of failure safety cases, TAGSI report TAGSI/P(97)140 Revision 6, 1998 (TRIM 2016/138079)*

[8] *EDF Energy, 'Assessment procedure for the high temperature response of structures,' Document R5, Issue 3, Revision 002, 2015*

[9] *EDF Energy, 'Assessment of the integrity of structures containing defects,' Document R6 Revision 4 Amendment 11, 2015*

[10] *British Standards Institution, 'Guide to methods for assessing the acceptability of flaws in metallic structures. Document BS7910:2013 and Amendment 1:2015*

[11] *Nuclear Installations Inspectorate, 'Statement on the operation of ferritic steel nuclear pressure vessels,' International journal of pressure vessels and piping, Vol. 64, No. 3, pp. 307-310, 1995*

[12] *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)*

[13] *'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)*

[14] *Harrop L P, 'Tables summarising International Atomic Energy Agency (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 (TRIM 2007/341485)*

[15] *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 (TRIM 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)
EMM	Enforcement management model
ESPN	Arrêté du 12 décembre 2005 relatif aux équipements sous pression nucléaires (French law on NPE, the ESPN Order)
EU	European Union
FANC	Federaal agentschap voor nucleaire controle (Nuclear safety regulator - Belgium)
HSE	Health and Safety Executive
IAEA	International Atomic Energy Agency
IPIRG	International Piping Integrity Research Group
ITPIA	Independent third party inspection agent
INSA	Independent nuclear safety assessment
IOF	Incredibility of failure
IRS	IAEA incident reporting system
LC	Licence condition
LOCA	Loss of coolant accident
MPI	Magnetic particle inspection
NDE	Non-destructive examination
ODPE	Piping failure data exchange (OECD) now subsumed by the Component operational experience, degradation and aging program (CODAP)
OECD	Organization for Economic Cooperation and Development

ONR	Office for Nuclear Regulation
ppm	Parts per million
PSA	Probabilistic safety analysis
PSI	Pre-service inspection
PWR	Pressurized water reactor
QA	Quality assurance
R2P2	Reducing risk protecting people
RPV	Reactor pressure vessel
RSRLs	WENRA reactor safety reference levels
SAP	Safety assessment principle
SSC	Structure, system or component
TAG	Technical assessment guide
TAGSI	Technical advisory group structural integrity
TECDOCS	Technical documents series (IAEA)
TEPCO	Tokyo Electric Power Company, Inc.
WENRA	Western European Nuclear Regulators Association
WSFSSRL	WENRA waste and spent fuel storage safety reference level

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

A1. 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

A1.1. 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.

A1.2. Generally the WENRA RSRLs are at a high level, somewhat similar to the higher level requirements of the SAPs. Rather than compare the RSRLs and the SAPs and show how they are similar, here, the differences relevant to structural integrity are highlighted. If the inspector applies the SAPs, the RSRLs should be accounted for. For example, the RSRL on ageing and degradation is covered in this TAG in §5.93 through §5.99. A few RSRLs are quite specific, relevant examples for structural integrity are *inter alia*:

- ***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 by all countries; see §5.47 & §5.60 of this TAG.

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 §104 & §105 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 (TRIM 2015/390992)*

[A1-2] *Western European Nuclear Regulators Association (WENRA), WENRA decommissioning safety reference levels,' 2011 (TRIM 2015/390989)*

[A1-3] *Western European Nuclear Regulators Association (WENRA), WENRA waste and spent fuel storage safety reference levels,' 2011 (TRIM 2015/390995)*

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9. APPENDIX A2: IAEA STANDARDS, GUIDANCE AND DOCUMENTS

A2. This appendix considers the implications of the IAEA documents for advice in the TAG. The documents are available on the IAEA website.

The documents

A2.1. 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 organizations to their own operations and are recommended for use by states and national authorities and by other international organizations 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.

A2.2. The result of this review of IAEA documents can be summarized 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;

A2.3. The lists of IAEA documents have been assessed and the subset that is potentially relevant to assessment of metal SSCs extracted and provides the references in this appendix.

A2.4. 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

A2.5. 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.

A2.6. 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.

A2.7. 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

A point-by-point comparison between the IAEA SSR-2/1 [A2-2] requirements and the SAPs would be unduly cumbersome; for example, §6.14 of SSR-2/1 states:

> *'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.'*

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

A2.8. 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 of 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 and 310.

A2.9. 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.

A2.10. 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.

A2.11. 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).

A2.12. 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. This is a relatively specialized area. 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.

A2.13. 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.

A2.14. NS-G-2.3 [A2-7] covers modifications. The guide is quite general; there is nothing in NS-G-2.3 which is only relevant to integrity of metal SSCs. 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. A modification to a safety case, in the light of new knowledge, but containing arguments why no other modifications are needed could be something ONR would wish to assess. Active management and maintenance of safety cases—which implies the potential for modifications to safety cases—is addressed in SAP SC.7. For 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 appropriate categorisation—the SAPs deal implicitly with modifications through the principles relevant to safety cases (SC.1 to SC.8).

A2.15. 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 to EHA.17. In terms of the integrity of metal SSCs, seismic loading is included (implicitly) in EMC.7.

Safety reports series

A2.16. 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. Advice on the assessment of closure head studs is given in [A2-30 & A2-44].

References

Safety standards series

A selection of 10 relevant documents from the 169 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-2.6, 2009

Safety reports series

A selection of three relevant documents from the 82 listed on the IAEA website:

[A2-11] IAEA safety reports series no. 82: Ageing management for nuclear power plants: International generic ageing lessons learned (IGALL) 2015

[A2-12] IAEA safety reports series no. 81: Development of a regulatory inspection programme for a new nuclear power plant project, 2014

[A3-13] IAEA safety reports series no. 62: Proactive management of ageing for nuclear power plants, 2009

Technical reports series

A selection of four relevant documents from the 400 listed on the IAEA website:

[A2-14] IAEA technical reports series: Methodology for the management of ageing of nuclear power plant components important to safety TRS 338, 1992

[A2-15] IAEA technical reports series: Guidelines for application of the master curve approach to reactor pressure vessel integrity in nuclear power plants TRS 429, 2005

[A2-16] IAEA technical reports series: Plant life management for long term operation of light water reactors - principles and guidelines TRS 448, 2006

[A2-17] IAEA technical reports series: Neutron irradiation embrittlement of reactor pressure vessel steels TRS 163, 1975

Technical Documents (TECDOCs)

A selection of 20 relevant documents from the 1770 listed on the IAEA website:

[A2-18] IAEA-TECDOC-189 Fracture mechanics applications: implications of detected flaws, Winterthur 3-5 December 1975, 1976

[A2-19] IAEA-TECDOC-510 Status of advanced technology and design for water cooled reactors: heavy water reactors, 1989

[A2-20] IAEA-TECDOC-677 Progress in development and design aspects of advanced water cooled reactors, 1992

[A2-21] IAEA-TECDOC-682 Objectives for the development of advanced nuclear plants, 1993

[A2-22] IAEA-TECDOC-752 Status of advanced containment systems for next generation water reactors, 1994

[A2-23] 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

[A2-24] IAEA-TECDOC-936 Terms for describing new, advanced nuclear power plants, 1997

[A2-25] IAEA TECDOC-968 Status of advanced light water reactor designs 1996, 1997

[A2-26] 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

[A2-27] IAEA-TECDOC-981 Assessment and management of ageing of major nuclear power plant components important to safety: Steam generators, 1997

[A2-28] IAEA-TECDOC-1037 Assessment and management of ageing of major nuclear power plant components important to safety: CANDU pressure tubes, 1998

[A2-29] IAEA-TECDOC-1119 Assessment and management of ageing of major nuclear power plant components important to safety - PWR vessel internals, 1999

[A2-30] IAEA-TECDOC-1120 Assessment and management of ageing of major nuclear power plant components important to safety - PWR pressure vessels, 1999

[A2-31] *IAEA-TECDOC-1181 Assessment and management of ageing of major nuclear power plant components important to safety: Metal components of BWR containment systems, 2000*

[A2-32] *IAEA-TECDOC-1197 Assessment and management of ageing of major nuclear power plant components important to safety: CANDU reactor assemblies, 2001*

[A2-33] *IAEA TECDOC-1263 Application of non-destructive testing and in-service inspection to research reactors - Results of a coordinated research programme, 2001*

[A2-34] *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*

[A2-35] *IAEA-TECDOC-1341 Extreme external events in the design and assessment of Nuclear Power Plants, 2003*

[A2-36] *IAEA-TECDOC-1347 Consideration of external events in the design of nuclear facilities other than nuclear power plants, with emphasis on earthquakes, 2003*

[A2-37] *IAEA-TECDOC-1361 Assessment and management of ageing of major nuclear power plant components important to safety: Primary piping in PWRs, 2003*

[A2-38] *IAEA-TECDOC-1390 Construction and commissioning experience of evolutionary water cooled nuclear power plants, 2004*

[A2-39] *IAEA-TECDOC-1391 Status of advanced light water reactor designs 1994, 2004*

[A2-40] *IAEA-TECDOC-1400 Improvement of in-service inspection in nuclear power plants, 2004*

[A2-41] *IAEA-TECDOC-1435 Application of surveillance programme results to reactor pressure vessel integrity assessment: Results of a coordinated research project 2000-2004, 2005*

[A2-42] *IAEA-TECDOC-1441 Effects of nickel on irradiation embrittlement of light water reactor pressure vessel steels, 2005*

[A2-43] *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*

[A2-44] *IAEA-TECDOC-1470 Assessment and management of ageing of major nuclear power plant components important to safety: BWR pressure vessels, 2005*

[A2-45] *IAEA-TECDOC-1471 Assessment and management of ageing of major nuclear power plant components important to safety: BWR pressure vessel internals, 2005*

[A2-46] *IAEA-TECDOC-1474 Natural circulation in water cooled nuclear power plants - phenomena, models and methodology for system reliability assessments, 2005*

[A2-47] *IAEA-TECDOC-1487 Advanced nuclear plant design options to cope with external events, 2006*

[A2-48] *IAEA-TECDOC-1503 Nuclear power plant life management processes: Guidelines and practices for heavy water reactors, 2006*

[A2-49] *IAEA-TECDOC-1556 Assessment and management of ageing of major nuclear power plant, 2007*

[A2-50] *IAEA-TECDOC-1557 Assessment and management of ageing of major nuclear power plant components important to safety: PWR vessel internals, 2007*

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

A3. 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 reactor pressure vessel steels
- Macrosegregation in reactor pressure vessel steels
- Sizewell A boiler shell cracking

Hydrogen-induced cracking in reactor pressure vessel (RPV) steels [A3-1]

A3.1. ***Plant history and 2012 outage findings***—Doel 3 and Tihange 2 are pressurized water reactor (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 reactor pressure vessel 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.

A3.2. ***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).

A3.3. ***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 reactor pressure vessel (RPV) head and bottom dome

A3.4. ***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 which are stipulated in France by the ESPN Order of 2005 [A3-8].

A3.5. ***Technical qualification***—The ESPN Order imposes a requirement for Technical Qualification. The objectives of the 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.

A3.6. ***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].

A3.7. ***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].

A3.8. ***Implications***—At the time of issue of this TAG, work on a justification for use of the Flamanville 3 RPV domes was under way. ASN had stated that while it accepted this approach, the outcome of the process was uncertain; ASN had to decide whether the domes were acceptable for service. 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

A3.9. ***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 reactor pressure vessel (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].

A3.10. ***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 (coded 2A, 2C and 2D) 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].

A3.11. ***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 development (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 parameters and processes to prevent cracking (EMC.5 & EMC.14)
- Avoidance or mitigation of damage to primary circuit and boiler components (EMC.5 & EMC.19)

A3.12. ***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.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 (§A3.11).

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11. APPENDIX A4 : CITATIONS OF SAPS FOR METAL SSCS AND FOR AGEING

Safety case area	Subject	Identity	Section
Highest reliability components and structures	Safety case and assessment	EMC.1	5.1, 5.12, A2.7, A3.11, A3.12
	Use of scientific and technical issues	EMC.2	5.1, 5.12, 5.13, A3.11
	Evidence	EMC.3	5.1, 5.12, 5.14, A3.12
General	Procedural control	EMC.4	5.26, 5.56, A3.11
	Freedom from and tolerance of defects	EMC.5	5.29, 5.57, 5.74, A2.7, A3.11
	Means to identify defects	EMC.6	5.57, 5.74, A2.7
Design	Loadings	EMC.7	5.27, 5.37, 5.48, A2.15
	Providing for examination	EMC.8	5.57
	Product form	EMC.9	5.29, 5.57
	Weld positions	EMC.10	5.29
	Failure modes	EMC.11	5.27, 5.37, 5.48, A2.7
	Brittle behaviour	EMC.12	5.45, A2.7
Manufacture and installation	Materials	EMC.13	5.53
	Techniques and procedures	EMC.14	5.56, 5.58, A3.11
	Control of materials	EMC.15	5.56
	Contamination	EMC.16	5.56
	Third-party inspection	EMC.18	5.58, 5.72, A3.11
	Non-conformances	EMC.19	5.68, A3.11
	Records	EMC.20	5.70
Operation	Safe operating envelope	EMC.21	A2.8
	Material compatibility	EMC.22	A2.8
	Ductile behaviour	EMC.23	5.45, A2.7, A2.8
Monitoring	Operation	EMC.24	A2.13
	Leakage	EMC.25	5.91, A2.13
	Forewarning of failure	EMC.26	5.91, A2.7, A2.13
Manufacturing, pre-service and in-service examination and testing	Examination	EMC.27	5.75, A2.13
	Margins	EMC.28	5.75, A2.13
	Redundancy and diversity	EMC.29	5.76, A2.13
	Qualification	EMC.30	5.76, A2.13, A3.11
In-service repairs and modifications	Repairs and modifications	EMC.31	A2.14
Analysis	Stress analysis	EMC.32	5.30, 5.40
	Use of data	EMC.33	5.33, 5.48
	Defect sizes	EMC.34	5.39, 5.40, A2.7
Ageing and degradation	Safe working life	EAD.1	5.26, 5.96, A2.13
	Lifetime margins	EAD.2	5.97, A2.13
	Periodic measurement of material properties	EAD.3	5.54, 5.97, A2.13
	Periodic measurement of parameters	EAD.4	5.26, 5.97, A2.13
	Obsolescence	EAD.5	5.98, A2.13