Office for Nuclear Regulation

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GRAPHITE REACTOR CORES			
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1. INTRODUCTION

1.1 The Office for Nuclear Regulation (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 dutyholders. 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 judgements and decisions. This technical assessment guide (TAG) is one of these guides.

2. PURPOSE AND SCOPE

2.1 The purpose of this TAG is to provide additional guidance and interpretation; and explain the application of ONR's Safety Assessment Principles for Nuclear Facilities, 2014 Revision 0 (SAPs) [1], for graphite reactor cores, principally for operating reactors and, to a lesser extent, for reactors during decommissioning.

2.2 Inspectors will need to use this TAG in combination with their existing experience and discussions with colleagues in ONR to understand aspects of the ONR assessment process in this area.

2.3 The outcome of an assessment is predominantly a consequence of the inspector's regulatory judgment and discretion, within the framework of ONR's assessment process. Assessment of graphite cores should not be undertaken in isolation, as there are other TAGs which are relevant in particular cases; these are listed in Section 5.20.

2.4 This TAG considers the applicability of the 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. WENRA and IAEA documents are considered in Appendices A1 and A2 respectively.

3. RELATIONSHIP TO LICENCE AND OTHER RELEVANT LEGISLATION

3.1 The primary licence conditions (LCs) [2] for which assessments of the safety of graphite components and cores are to be carried out are:

- LC 15 (Periodic review)
- LC 21 (Commissioning)
- LC 22 (Modification or experiment on existing plant)
- LC 23 (Operating rules)
- LC 24 (Operating instructions)
- 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 35 (Decommissioning)

3.2 Other licence conditions relevant to assessment of graphite components and cores are:

- LC 6 (Documents, records, authorities and certificates)
- LC 10 (Training)
- LC 12 (Duly authorised and other suitably qualified and experienced persons)
- LC 13 (Nuclear safety committee)
- LC 14 (Safety documentation)
- LC 17 (Management systems)

- LC 18 (Radiological protection)
- LC 25 (Operational records)
- LC 31 (Shutdown of specified operations)
- LC 32 (Accumulation of radioactive waste)
- LC 33 (Disposal of radioactive waste)
- LC 36 (Organisational capability)

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

4.1 The ONR Safety Assessment Principles (SAPs) EGR.1 to EGR.15 (paragraphs 365 to 394) directly address the assessment of graphite reactor cores [1]:

- Safety cases
  - EGR.1 (Safety cases)
- Design
  - EGR.2 (Demonstration of tolerance)
  - EGR.3 (Monitoring)
  - EGR.4 (Inspection and surveillance)
- Manufacture, construction and commissioning
  - EGR.5 (Manufacturing records)
  - EGR.6 (Location records)
- Component and core condition assessment (CCCA)
  - EGR.7 (Materials properties)
  - EGR.8 (Predictive models)
  - EGR.9 (Materials property data)
- Defect tolerance assessment (DTA)
  - EGR.10 (Effects of defects)
  - EGR.11 (Safe working life)
  - EGR.12 (Operational limits)
  - EGR.13 (Use of data)
- Monitoring
  - EGR.14 (Monitoring systems)
- Examination, inspection, surveillance, sampling and testing (EISST)
  - EGR.15 (Extent and frequency)

Appendix A3 indicates the sections of this TAG that cite these SAPs. Of the other SAPs, those on ageing and degradation are particularly relevant to graphite reactor cores:

- 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 Other relevant SAPs include:

- AV.1 to AV.8 (Assurance of validity of data and models)
- DC.1 to DC.9 (Decommissioning)
- ECE.12, ECE.14, ECE.17 (Civil engineering)
- ECS.1 to ECS.5 (Safety classification and standards)
- EHA.1-7, EHA.9, EHA.18 (External and internal hazards)
- EKP.1 (Key principles)
- ELO.1 (Layout)
- EMC.3 to EMC.8, EMC.11, EMC.13 to EMC.22, EMC.24 to EMC.30, EMC.32 to EMC.34 (Integrity of metal components and structures)
- EMT.1 to EMT.8 (Maintenance, inspection and testing)
- ERC.1-4 (Reactor cores) — this set of principles are applicable to all reactor core types. They relate to the need to control reactivity, heat generation / removal and other aspects of the design so that components within the reactor can be kept within specified limits to ensure an appropriate level of safety during operation and in design basis fault conditions. Subsequently, these reactor core principles are also applicable to graphite reactor cores in addition to the specific principles for graphite cores mentioned above, EGR.1-15
- ERL.1 and ERL.2 (Reliability claims)
- FA.1 to FA.16 and FA.25 (Fault analysis)
- MS.2 and MS.4 (Leadership and management for safety)
- RW.1 to RW.7 (Radioactive waste management)
- SC.1-8 (Safety case characteristics)

WENRA Reference Levels and IAEA Safety Standards

4.3 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. The WENRA and IAEA documents are for nuclear reactor power plants but they provide lesser guidance and requirements relating to graphite-specific issues than the SAPs and this TAG.

5. ADVICE TO INSPECTORS

Introduction

5.1 This guide provides advice on the assessment of safety cases for graphite cores using the approach set out in the SAPs.

5.2 All operating UK gas-cooled reactors have graphite cores. In addition, a number of reactors with graphite cores or reflectors are being decommissioned and irradiated graphite waste is stored at facilities at various sites. As stated in §2.1, this TAG applies principally to graphite cores of operating reactors and to a lesser extent to graphite cores of reactors during decommissioning. However, the knowledge provided could be applied to reflector graphite components or stored graphite components if the necessity arises.

Graphite reactor cores safety case requirements

5.3 The safety case for a graphite reactor core should:

- Identify the structures, systems and components that are important for safe operation.
- Identify normal operating and potential fault conditions, including the effects of internal and external hazards.
- Demonstrate that the integrity of structures, systems and components important for safe operation is maintained for a defined period of operation. Ultimately this will be the projected life of the installation, including any period of safe storage and decommissioning, taking due account of ageing and degradation

mechanisms. In service, these may affect the safety functions of graphite cores by:

- Changing the size, shape and position of graphite components
- Changing their properties - including stored (Wigner) energy
- Developing internal stresses
- The initiation and growth of cracks
- Developing forces and moments between components
- Forming potentially mobile debris

5.4 Additionally, there are aspects of the functionality of graphite components and cores for which structural integrity may not be sufficient (*e.g.* oxidation, which *inter alia* reduces moderation).

5.5 It is important to be aware of the functionality aspects of a graphite cores safety case within the broader safety case; EGR.1 refers to the need for the safety case to demonstrate either that a graphite core is free from defects that could impair its safety functions or that the safety functions are tolerant of defects.

5.6 The inspector should consider the adequacy of margins against failure conditions, inspection capabilities, and integrity analysis in the context of an entire safety case taking due note of relevant precedents. The margins should demonstrate clearly that the safety case limits will not be breached during any justified period of operation.

Treatment of faults and risk

5.7 In reaching a judgment on the adequacy of a graphite core safety case, the inspector needs to establish, in broad terms, the likely implications of the core failing to perform its safety functions and reach a judgment on the probability of their occurrence (ERL.1). For example, a severe radioactive release accident could result from the core impairing the insertion of the control rods if the reactor trips and the reactor cooling systems have changed to their post-trip cooling modes which are sufficient only to carry away decay heat.

5.8 ERL.1 paragraph 191 indicates that for some structures, systems or components, adequate reliability data may not be available. The graphite core is in this category. In such cases, a considered judgment should be made of the contribution to the predicted frequencies from such faults. The inspector—in reaching a judgment on the adequacy of the safety case—should consider the contribution of core related initiating faults and seek advice from probabilistic safety analysis (PSA) and fault studies inspectors on the impact of the PSA and fault sequences (paragraph 656(c)).

5.9 FA.5 paragraph 628 requires that the safety case present a list of all initiating faults which are included within the design basis of the plant. All initiating faults should be considered, but failures of components or systems, for which acceptable case-by-case arguments have been made in accordance with ERL.1 paragraph 191 need not.

5.10 SC.3 requires that the control of the hazard should be demonstrated by a valid safety case that takes account of the implications for future lifecycle stages. Hazard is defined in the SAPs Paragraph 9 as 'anything that is capable of causing harm is termed a “hazard”'. Damaged graphite components have the potential to impair or result in failure of the core to passively perform its safety functions. In this respect, graphite component damage, such as brick cracking, should be treated as a hazard, or a precursor to an initiating event.

5.11 For a graphite core safety case, the inspector needs to consider—for each component and damage mechanism—the possible effects of ageing and at which plant states and stages of the life of a reactor they might affect safety. Since the probabilities of such challenges to safety are likely to increase with time, it is difficult to assign precise failure probabilities. The starting point is to identify damage mechanisms and how these may affect core safety functions. Conservative estimates of the probabilities of core safety functions being affected, over defined periods, are required as is a consideration of consequences. Inspection and monitoring will be necessary to ensure that the estimates are conservative.

5.12 Any new core condition or operating regime needs careful consideration; prior acceptable behaviour should not necessarily be used as a guide to future performance. Factors to consider are: the operating environment and uncertainties in operating and fault conditions, physical data and design methods (ERL.1). Commonality arguments may offer some assurance of component and core reliability when operating experience is available from a nominally identical core with damage that bounds that of a core of interest. However, uncertainties in as-manufactured condition, effects of exposure conditions and fault conditions should be considered. Where appeal is made to the operating experience of a different graphite component or core—if that core has not experienced similar operating conditions, transients or faults that could affect component or core damage, and hence likelihood of challenges to safety functionality—care needs to be exercised in the application of reliability and performance data (ERL.1). Appropriate measures should be taken to ensure that the onset of component and core damage can be detected, and that the consequences of such damage are acceptable and minimized.

5.13 The SAPs Targets 8 and 9 give quantitative expression to the concept of an indicative upper limit on risk, the Basic Safety Levels (BSLs) and an upper limit on the broadly acceptable region, the Basic Safety Objectives (BSOs). It is not always possible to apply such quantification, and instead assessment will be more in terms of qualitative likelihood (FA.13 paragraph 656(c)).

5.14 If it seems that a BSL is exceeded, the inspector should read SAPs paragraphs 698 to 700. The issue should be considered using the ONR enforcement management model (EMM). Inspectors should consult Procedure NS-ENF-GD-002 [3] and Guide ONR-ENF-GD-006 [4] for advice on use of the EMM.

5.15 If a safety case is not in the 'broadly acceptable' region (i.e., below the Basic Safety Objectives (BSOs)), the assessment of the risk is based on balancing the strengths and weaknesses in the safety case which inform a judgment as to whether the risk is at or above the limit of tolerability (i.e., above the Basic Safety Limits (BSLs)). This judgment should consider the safety functional requirements of the graphite cores and to what extent they could be compromised.

5.16 In situations where the risk is below the limit of tolerability, the case may be adequate to secure operation in the short term while ALARP improvements are made. There is no one set of minimum conditions to judge achievement of the limit for tolerability, but inspectors should give due consideration to the threat to the following safety functions and seek advice from fault studies inspectors:

- Safe shutdown and hold-down (ERC.2)
- Maintenance of component and core geometry to secure fuel cooling (ERC.3)
- Removal of fuel (ERC.3)
- Monitoring of safety related conditions (ERC.4)

5.17 EHA.1 and EHA.19 require, for all hazards, that the design basis analysis principles and the PSA principles are satisfied as appropriate, unless it can be demonstrated that:

- The frequency of an event being exceeded is less than once in ten million years, or

- The impact of the hazard has no effect on the safety of the facility

These exclusions do not apply if the frequency of realisation and the potential impact might make a significant contribution to the overall risks from the facility. Therefore—unless it can be demonstrated that the frequency of the core failing to perform its passive safety functions is essentially incredible—in assessing the safety case, the inspector should consult colleagues to investigate the impact of the core on overall plant risk.

5.18 A graphite core safety case should be examined in the context of the overall safety case for the plant, taking due account of interactions with other safety features (EGR.1 paragraphs 370 and 371). There may be protective devices that can mitigate effects of core damage; alternatively, the direct effect of failure may be trivial but the consequences may be significant (*e.g.* damage to safety related plant or operator dose uptake).

Safety case approaches for graphite reactor cores and those for metal components

5.19 There are similarities between the assessment approach in this TAG and that in the TAG for the integrity of metal components and structures [5]. For instance, a requirement throughout the assessment process is that analytical models should use methods that have been verified and validated in accordance with EGR.1, for graphite core safety cases, paragraph 374. However, there are additional issues which need to be considered for graphite reactor cores:

- Potential interactions between ageing mechanisms
- Limited knowledge of ageing mechanisms
- Substantial potential for interaction between components, particularly in the longer term
- Limited understanding of crack initiation and growth mechanisms
- Redundancy in some aspects of functionality

Thus an aspect of the assessment of a graphite core safety case is to gain an assurance that several safety functions are maintained. Graphite core safety cases may cover situations where integrity is assured or where there is tolerance to local failure, which may be reflected in additional legs in a safety case to inform judgments on the threat to the safety functions where there is redundancy. In contrast, much of the guidance in the TAG for the integrity of metal components and structures [5] concerns the structural integrity of non-redundant components.

Relevant TAGs

5.20 The following TAGs may be relevant:

- NS-TAST-GD-005 Demonstration of ALARP (As low as reasonably practicable)
- NS-TAST-GD-009 Examination, inspection, maintenance and testing of items important to safety
- NS-TAST-GD-013 External hazards
- NS-TAST-GD-014 Internal hazards
- NS-TAST-GD-024 Management of radioactive materials and radioactive waste on nuclear licensed sites
- NS-TAST-GD-026 Decommissioning
- NS-TAST-GD-030 Probabilistic safety analysis
- NS-TAST-GD-033 Duty holder management of records
- NS-TAST-GD-035 Limits and conditions for nuclear safety (operating rules)
- NS-TAST-GD-041 Criticality safety
- NS-TAST-GD-042 Validation of computer codes and calculation methods
- NS-TAST-GD-050 Periodic safety reviews (PSR)

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

Safety functions of graphite cores

5.21 The graphite cores provide:

- Neutron moderation
- Neutron reflection which enhances neutron efficiency and provides shielding
- Passages for the entry and movement of control rods and fuel stringers
- Mass
- Channels that direct the flow of coolant

The (related) safety functions are to:

- Enable shutdown and reactivity control post shutdown
- Allow fuel and core cooling functions to work during operation, transients, faults and post shutdown
- Avoid challenges to fuel integrity through core physical changes and responses
- Enable removal of fuel from the reactor
- Enable reactivity control during operation and under fault conditions
- Provide thermal inertia during transients, faults and post-shutdown
- Provide weight to hold down the cores and gas baffles

Since the graphite cores form a principal means of fulfilling these nuclear safety functions, their categorisation will be Category A and Class 1 (ECS.1 and ECS.2).

For graphite cores under decommissioning, the applicable purposes and safety functions of the core will reduce depending on the decommissioning stage (see the Decommissioning section of this TAG).

5.22 With regard to moderation and possible reactivity faults, ERC.1 paragraph 543 states:

> *'No single moveable fissile assembly, moderator or absorber when added to or removed from the core should increase the reactivity by an amount greater than the shutdown margin, with an appropriate allowance for uncertainty. The uncontrolled movement of reactivity control devices should be prevented.'*

The effects of oxidation on graphite core density for steam ingress faults should be considered in this context. Graphite weight loss will affect reactivity and—for under-moderated cores—erode margins for shutdown, hold-down and for reactivity control.

5.23 For reactor cores where the upwards force caused by pressure across the top box dome is resisted by the weight of the attached components—mostly that of the graphite core—weight loss will reduce the pressure at which the assembly could levitate, challenging core alignment and ability to ensure shutdown and hold-down.

5.24 Graphite core safety depends on a number of inter-related technical areas, including: reactor physics, materials science (including the physical and mechanical responses of graphite to irradiation), the chemical interactions between graphite and reactor coolant gas, thermo-hydraulics, stress analysis, whole core modelling and inspection techniques. While an inspector may be experienced in a number of these, the inspector should be alert to those aspects of an assessment where there is a need to consult with colleagues or seek advice external to ONR.

Multi-legged graphite core safety cases

5.25 Graphite core safety cases usually need to adopt a multi-legged approach. This section discusses possible legs and how an inspector may judge their contributions to an overall safety case.

5.26 EGR.1 paragraph 369 identifies which aspects should be included in a graphite core safety case (Reference [6] presented a framework for judging the adequacy of a graphite core safety case and gave examples of precedent from previous assessments):

- Design
- Manufacture, construction and commissioning
- Component and core condition assessment (CCCA)
- Defect tolerance assessment (DTA)
- Analysis of radiological consequences of defects
- Monitoring
- Examination, inspection, surveillance, sampling and testing (EISST)

5.27 Where a multi-legged safety case is possible and the legs of the case are independent, a strong leg of the safety case may offset a weakness elsewhere. The strength of each leg may vary through life and ALARP improvements include consideration of how increased confidence may be achieved and demonstrated from each leg of the safety case. Thus the balance of the safety case arguments and judgments as to whether the risks are ALARP are likely to differ for new and existing plant.

5.28 In considering a safety case that appears to be multi-legged, the inspector needs to establish whether each leg is independent. The most robust case would be one where all legs were independent with redundancy and diversity in the arguments presented. If more than one leg is underpinned by common data, assumptions or methodologies, then the strength of a multi-legged case may be undermined. For example, the DTA leg may use results from the CCCA leg. In such a case, the independent leg needs to demonstrate a strong argument and/or the dependent leg should demonstrate tolerability to the uncertainties associated with the independent leg.

5.29 Strengths and weaknesses in different legs may offset one another. For example, as a consequence of limited data for relevant ageing and degradation mechanisms and incomplete validation of methods, the CCCA leg could be weak and there would need to be greater reliance on other legs, possibly the inspection leg with limited extrapolation of data. In that case, it would be important to have arguments to show that any degradation in safety functionality were gradual over the proposed period of operation. Moreover, a monitoring leg that demonstrated forewarning of an accelerated degradation processes would be a valuable addition to such arguments.

Design

5.30 Since the designs of the graphite cores of the UK gas-cooled reactors were not based on any recognised published design code or standard, the inspector should establish that a safety case is based on sound scientific understanding and includes a comprehensive examination of all the relevant scientific and technical issues supported by suitable experimental verification and validation (ERL.1 paragraph 191).

5.31 The safety case for graphite components and structures which form a principal means of ensuring nuclear safety should include evidence of sound design concepts and proven design features, EMC.3 paragraph 295(a). ERC.1 requires that a reactor core design takes account of all:

- Operating modes including normal operation, refuelling, testing, shutdown and fault conditions
- Identifiable environmental effects including irradiation, chemical and physical processes
- Static and dynamic mechanical loads, thermal distortion and thermally-induced stress
- Possible variations in manufacture

and any other identified safety-related factor. The design should demonstrate tolerance of the core safety functions to ageing processes and imposed loads (EGR.2).

5.32 The design should include:

- Monitoring systems to enable the core to be maintained within its safe operating envelope throughout the life of the installation (EGR.3)
- Features to enable inspection during manufacture and service and for graphite surveillance samples (EGR.4)

Manufacture, construction and commissioning

5.33 For graphite cores which form a principal means of ensuring nuclear safety, the safety case should include evidence to demonstrate that the necessary level of integrity has been achieved for the most demanding situations (as for EMC.3 for the highest reliability metal components and structures).

5.34 For graphite components (particularly graphite sleeves) the specification of a proof test before service provides some assurance that the as-built structure has been constructed to an adequate standard (EMC.17). That is the material strength and section thicknesses are adequate.

5.35 Care is required in accepting commonality arguments based on manufacture, operational experience or inspection 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.

5.36 The standard of design and construction of graphite components should be appropriate to their safety classification (ECS.2 and ESC.3). The inspector may need to examine a case history to confirm that it contains adequate records of the specification of detailed component design, in-process inspection, testing, inspection procedures and location (EGR.5 and EGR.6). However, construction records do not always show the full picture, and it may be appropriate to consider other options such as inspection.

Component and core condition assessment (CCCA)

5.37 The component and core condition assessment (CCCA) leg of a graphite core safety case will predict the condition of components and cores at a defined stage in life. It will comprise:

- Predictions of graphite component properties and condition (EGR.7) from an assessment of:
  - Field variables: stress, neutron dose, gas composition, weight loss and temperature
  - Materials properties data with allowances for irradiation and oxidation
- Predictions of core condition from an assessment of:
  - Component condition and interactions between graphite core components and other items including fuel, control rods, core support structures, restraint structures and monitoring devices.
  - Normal operation, transients, faults and hazards

5.38 The assessment will need to make appropriate use of whole core, or partial core models, to predict complex component-core interactions (EGR.7 paragraph 380) and consider all reasonably foreseeable damage and failure mechanisms.

5.39 The assessment should consider stochastic and systematic effects and, in doing so, will need to investigate the likelihood of damage clusters or damage cascades.

5.40 The assessment may be deterministic with sensitivity studies and/or probabilistic to allow for variability in graphite behaviour and other uncertainties. Ideally, the probabilistic study should provide support in depth to the deterministic study by demonstrating clear safety margins and conservatism.

5.41 The inspector may wish to consider the following aspects of the CCCA leg:

- Information on the graphite properties from:
  - Tests on the as-manufactured graphite
  - Samples removed from operating reactors
  - Experiments in materials test reactors (MTRs)
- The adequacy of mechanistic understanding of graphite material's behaviour
- Information on how the cores were constructed. (Ideally, the location of each brick, and its associated properties would be known.)
- The extent to which the methodology is validated
- The validation of failure criteria
- The understanding of cracking behaviour
- Conservatism or accuracy of predictions of component and core behaviour

5.42 The following ageing processes and their potential interactions should be considered in any prediction:

- Radiolytic oxidation
- Effects of neutron dose
- Irradiation creep
- Onset of cracking and cracking rate
- Damage (or crack) progression
- Geometrical shape changes of core components

5.43 AV.2 states that calculation methods [models] used for the analyses should adequately represent the physical and chemical processes taking place. For the CCCA leg, many models are used to predict graphite properties and field variables (as listed in §5.37 and subject to the ageing process listed in §5.42).

5.44 Any such models should be based on a sufficient and sound scientific understanding and any necessary assumptions or approximations should demonstrably bias results in a safe direction (AV.2 and EGR.8). Models or methods that appear to be conservatively biased for one mechanism or mode of core damage may not be so for others and—because of the complexity of some models—this may not be immediately apparent. Under such circumstances, sensitivity studies may be appropriate.

5.45 AV.3 and EGR.8 require that analytical models be validated against experiments that replicate the expected plant condition as closely as possible. Care is cautioned in interpretation and application of analytical models to anticipated plant conditions to allow for uncertainties. Where appropriate, results from a model should be checked against independent predictions.

5.46 Provision should be made to keep under review: new data, scientific knowledge, research and operating experience to ensure that a safety case is not invalidated (SC.7 paragraphs 108 and 110).

5.47 TAG NS-TAST-GD-042 provides guidance on the validation of computer codes and calculation methods. Predictive models should be shown to be valid for the particular application and circumstances by reference to established physical data, experiment or other means. Where uncertainty exists in these models appropriate safety margins should be demonstrated to account for this uncertainty. EGR.9 calls for extrapolation and interpolation from available data to be undertaken with care and robust justification provided for data and model validity beyond the limits of current knowledge.

5.48 EMC.7 and EMC.11 for metallic components are applicable. All operational loadings and credible fault loadings should be identified and their magnitudes specified. Failure modes should be gradual and predictable to avoid sudden step changes in the core behaviour. Load combinations should be defined. EHA.1, EHA.3 to EHA.7 cover external and internal hazard loads. Load definitions should be conservative, and remain appropriate for proposed future operation. This is of particular importance when assessing proposals for life extension. EHA.5 states that analysis of design basis events should assume the event occurs simultaneously with the facility's most adverse permitted operating state.

5.49 The assessment 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 life (EMC.32). Consideration should be given to the uncertainties associated with environmental loading when reliance is placed on out-of-reactor testing. The adequacy of margins against failure needs to be considered in the light of the accuracy, reliability and conservatism of analysis and test results, their scatter and areas of uncertainty.

5.50 The inspector should establish that data and assumptions used in the analysis are soundly based and demonstrably lead to conservative outcomes. Where multiple relevant data sets exist, consideration should be given to using the data that is most conservative to the safety case. Appropriate studies should be carried out to establish the sensitivity to the analysis parameters (EGR.13) and to identify any potential cliff-edge effects on the safety case. The uncertainties associated with material properties affected by degradation should be taken into account (EMC.33). 'Reducing Risk Protecting People' (R2P2) [7] is recommended for guidance and precautions in the face of uncertainty.

5.51 EAD.2 paragraph 215 requires that a design should allow for any uncertainties in determining the initial state of components and the rate of age-related degradation of components in the core environment. For a graphite core, this is particularly important as the number of graphite components is typically many thousands and—because of the nature of the graphite production processes—significant variability in as-manufactured material properties may be expected. The inspector should be aware that the extent of variability in the graphite materials may be size dependent.

5.52 For infrequent events, the inspector may need to consider whether there is scope for alleviation of the most rigorous requirements for pessimism in the analysis. In such cases, the safety case should provide a suitable justification for any relaxation. It may be reasonable to have stress limits that increase as the likelihood of loading decreases. That is lower stress limits would apply to normal operating conditions, and higher stress limits to infrequent fault loading conditions. This should be allowed only if the safety functions of the core remain demonstrably functional with no cliff-edge effects.

5.53 For damage or deterioration where the radiological consequences of failure are significant, the safety case should be supported by detailed analysis to demonstrate that the core is capable of withstanding the identified normal operating and potential fault loads for the lifetime of the installation. This may include finite element stress analysis. Procedures should be adequately verified and validated for the particular application as required by EMC.32 to EMC.34 and paragraph 316. Model testing may be necessary to confirm the adequacy of such analyses.

Defect Tolerance Assessment (DTA)

5.54 The objective of the DTA is to establish the effects of defects on the delivery of the safety functions for graphite components and structures (EGR.10). The DTA would demonstrate the level of tolerability to defects, so operational limits can be set. It should take due account of potential degradation mechanisms, such as progressive cracking of graphite components due to ageing. Where there is an increasing likelihood of challenge to core safety functions with time, the burden of proof and the robustness of the safety case should increase to meet the challenge.

5.55 The likely alternative approaches for the DTA leg are:

- Use predictions from the CCCA analysis to establish whether the core's passive safety functions are tolerant to the predicted damage. This has the disadvantage that the same assumptions, data and analysis underpin both the CCCA and the DTA. This is termed a 'dependent DTA'.
- Assume a level of component and core damage to which core safety functions are demonstrated to be tolerant.
- Establish the limit of component and core damage beyond which core safety functions are no longer tolerant. A safety margin would then need to be applied to this limiting condition (this may be in terms of time or degree of damage) and demonstrate that the core is always within this safe operating envelope.

The inspector should be aware that both of the last two approaches may be underpinned by complex numerical whole and/or part core models whose input is based on output from the CCCA analysis (see §5.28).

5.56 From each of the above possible approaches examination, inspection, surveillance, sampling and testing (EISST) programmes may be developed to track proximity to the assumed condition. Such a programme should enable any limiting condition to be detected using statistically significant sample sizes (at a suitably high confidence level) before conditions arise that may challenge core safety functions or safety case assumptions. EGR.10 paragraph 384 advises that—where the DTA is unable to clearly demonstrate that safety functionality is achieved and demonstrated under all reasonably foreseeable conditions—it may be necessary to resort to a consequences case (refer to §5.64 to §5.69). The inspector should establish whether a case is a damage tolerance case or—where the core safety functions are not damage tolerant—that the case has addressed the consequences for graphite core safety.

5.57 There should be a margin between the operating and fault envelope and any assumed condition over the full intended lifetime with due allowance for uncertainty (EGR.11 paragraph 385). If component damage is shown, or assumed to occur, effects on core safety functions should be shown to be progressive, with the possibility of disruptive failures—without adequate forewarning—being remote and detectable. The DTA needs to consider local and global effects of component and core damage.

5.58 The inspector should establish that data and assumptions used in the analysis are soundly based and demonstrably lead to conservative outcomes. Where multiple relevant data sets exist, consideration should be given to using the data that is most conservative to the safety case. Appropriate studies should be carried out to establish the sensitivity of analysis to uncertainties of its parameters and to identify any potential cliff-edge effects on the safety case.

5.59 EGR.11 and EGR.12 require a demonstration that the core and its components can be operated and controlled within a safe operating envelope throughout its life. Following service exposure, the core must remain fit to perform its safety functions despite changes in geometry and the effects of degradation mechanisms. Parameters of the operating envelope should be consistent with the type of construction, potential modes of failure and operational considerations (EMC.21).

5.60 The DTA should demonstrate that the core is stable in normal operation and does not undergo sudden changes of condition when operating parameters go outside the specified range. The geometry should be maintained within limits that enable passage of sufficient coolant to remove heat from all parts of the core. Where appropriate, means should be provided to minimize the chance of any obstruction of the coolant flow that could lead to overheating (ERC.3 paragraph 547). As ageing and damage progress, in-service changes to component and core condition and geometry might reduce, divert or impede coolant gas flow and thus reduce the effectiveness of cooling. The inspector should ensure that the licensee has considered any mechanisms that might be significant and undertaken suitable and sufficient modelling of predicted or assumed conditions. Models of component, core and coolant behaviour should be adequately verified and validated. The degree of rigour should be commensurate with the safety significance of existing and predicted credible damage as well as the likelihood of potential damage.

5.61 Safe reactor shutdown and hold-down should not be inhibited by component and core damage under normal operation or fault conditions. The inspector should investigate—with a fault studies inspector—any damage mechanisms which may impede control rods and safe operation of any other shutdown or hold-down systems. Not impeding the insertion of control rods is a safety requirement of the IAEA Safety Standards [9] that should be met for all design basis events. For the purpose of this TAG, any delay in the insertion of a control rod due to component or core damage is considered "impeding".

5.62 Difficulties in removing fuel from fuel channels may occur as fuel and/or fuel channel distorts with irradiation. The ability to remove fuel safely from the core should still be maintained despite any environmentally induced damage such as bowing or other damage in normal operation and/or design basis fault conditions (ERC.4 paragraph 557). Inability to remove fuel by normal methods would represent a major change in the operation of the reactor. If necessary, fuel retrieval could be linked to plans for post operational clean out and defuelling (Stage 1 decommissioning [8]).

5.63 For advice on operational and fault loadings, see §5.48 (CCCA Assessment).

Analysis of radiological consequences of defects

5.64 Where the DTA leg is unable to clearly demonstrate that margins are large enough to cover residual uncertainties, the consequences for core safety functionality need to be assessed to demonstrate that there are no cliff edge effects and to provide further evidence of defence in depth beyond the DTA. This will need to include a consideration of the hazards, likelihood of affecting core safety functions and the consequences.

5.65 The consequences argument needs to assess the likelihood and consequences of component and core damage, for each damage mechanism, so that this information may be used in the fault studies to demonstrate the effectiveness of reactor protection or potentially by the PSA if the risk of fuel failure cannot be ruled out.

5.66 The consequences leg may be an independent, dependent or supportive leg. It is unlikely that a safety case could be made based on a consequences leg alone as it is unlikely that credible initiating event/fault frequencies could be generated especially where a consequences leg is developed to address damage caused by time-dependent ageing.

5.67 The assessment should consider the effect of the core on operability of the shutdown, hold-down and core cooling systems. The analysis undertaken needs to consider local and global damage, uncertainties and the resultant effects. Faults and hazards—such as boiler tube failures or depressurisation—which can cause consequential damage, need to be considered.

5.68 The inspector must be convinced that any interference between the control rods and the graphite will not prevent the insertion of control rods to execute their shutdown and long term hold-down safety functions or cause problems with the control of reactor power. It should also be demonstrated that there are no cliff-edge effects as a result of a control rod impairment to mitigate uncertainties (EHA.7).

Monitoring

5.69 Plant monitoring in this context is defined as the gathering of information on plant condition or operating parameters during operation. Monitoring may take the form of actual core safety functions such as freedom of control rod and fuel movements or surrogates.

5.70 The monitoring leg of a graphite core safety case is potentially a fully independent leg and, consequently, a strong leg. However, the inspector should establish whether the monitoring undertaken is a lead or lag indicator of component and core condition. If monitoring systems, processes and procedures indicate a safety function impairment before action can be taken (or the process requires a response), it would be a lag indicator. Alternatively if action can be taken to remedy a situation before safety function impairment occurs, then monitoring is a lead indicator. Clearly the latter is a stronger leg as the former only allows a response after defence-in-depth is challenged. Monitoring is likely to be essential unless the CCCA clearly and demonstrably shows that no component and core damage occurs during operation and even then some confirmatory monitoring would be expected.

5.71 Guidance on provision for monitoring and on the monitoring of safety-related parameters is given in SAPs EMT.6 and ERC.4 respectively.

5.72 The scope and extent of monitoring should be commensurate with the required reliability. Where monitoring cannot be undertaken measures should be taken to compensate for the deficiency. Alternatively it should be demonstrated that adequate component and core performance will be achieved without such measures (EMT.6).

5.73 Where enhanced monitoring is introduced (over and above that previously undertaken), the safety case should consider measurements made in the past to ensure that any history is considered that may not have been previously correctly interpreted, *i.e*. previous trends need to be considered in developing criteria against which to assess future behaviour.

5.74 EGR.14 paragraphs 389 to 392 state that monitoring should be performed at appropriate intervals or continuously to ensure that the results will enable timely identification of degradation. The inspector may need to establish that the licensee has adequate arrangements to identify suitable precursors to changes of core safety functionality so that suitable and sufficient criteria are available to warn of impending challenges; that reporting and acceptance criteria are developed; and that arrangements for the evaluation of monitoring results are adequate. Arrangements should be in place to secure timely responses to mitigate untoward trends.

Examination, Inspection, Surveillance, Sampling and Testing (EISST)

5.75 EGR.15 states that:

> *'In-service examination, inspection, surveillance and sampling should be*

of sufficient extent and frequency to give confidence that degradation of graphite reactor cores will be detected well in advance of any defects affecting a safety function.'

5.76 Inspection before and during service has three objectives:

- To help to confirm that the plant is in the configuration assumed in the safety case
- To help to confirm any predicted degradation or ageing effect is developing as anticipated and remains within the limits set in the safety case
- To help to confirm there is no manufacturing shortfall or in-service degradation process other than those dealt with in the safety case

5.77 An EISST programme should demonstrate that the existence of defects in a graphite core can be established throughout its operational life (EMC.6) to confirm predicted behaviour (including changes in materials properties) or detect any unexpected safety-related effects or trends.

5.78 For graphite cores, in-service examination and inspection is likely to form a significant element of the safety case, especially for aged reactor cores. Inspection provides an important element in establishing the integrity of components and structures that form a principal means of ensuring nuclear safety.

5.79 Given the large numbers of channels and bricks present, it may not be reasonably practicable to inspect statistically significant sample sizes for an entire reactor core. However, it may be reasonably practicable to sample statistically significant sample sizes from targeted locations. The inspector should bear the various options in mind when judging the adequacy of any EISST proposals and when seeking ALARP improvements. Moreover, the inspector should consider whether it is reasonably practicable to increase sample sizes where there is an increasing uncertainty, e.g., due to aging, or a weakness identified in the other legs of the safety case.

5.80 The requirements for in-service EISST, or other maintenance procedures, and frequencies for which specific claims are made in the safety case should be identified and included in the maintenance schedule (EMT.1).

5.81 Pre-service inspection (PSI) and in-service inspection (ISI) procedures should be adopted which ensure that the initial condition of graphite components and core configuration, and any service induced changes can be reliably detected to secure safety case claims for component and core reliability (EMT.5 and EMT.6). Such inspections should be of sufficient extent and frequency to give adequate confidence that degradation will be detected well in advance of any component damage developing into a degraded core condition and affecting core safety functionality.

5.82 Provision should be made to keep data from EISST under review to ensure that the safety case is not invalidated (SC.7 paragraphs 108 to 110).

5.83 The extent and periodicity of inspection should be commensurate with the operational duty, reliability and safety functional requirement (EMT.5). Where defects, degradation or deviations from design intent are found, any proposed remedial action or technical justification should be assessed via the Licensee’s management processes, including Independent Nuclear Safety Assessment (INSA). Planning the extent of in-service inspection based on previous experience 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 inspection to look for the unexpected. A good number of degradation phenomena have been found initially by simple visual inspection methods, rather than sophisticated inspection techniques.

5.84 It should be demonstrated that structures are inspected to appropriate standards (ECS.3), are defect tolerant and that the existence of defects can be established by inspection throughout the operational life (EMC.6).

5.85 In-service inspections should be carried out where they are reasonably practicable to enable the condition of the core to be confirmed, and to verify that the structure is behaving as the safety case assumes. In-service inspection provides a means of assuring that structures remain at all times fit for purpose (EMC.27), and checks for the unexpected.

5.86 Difficulties may arise in interpreting re-inspection results where modifications have been made to the inspection procedures following the original inspections.

5.87 Inspection procedures and inspection personnel should be qualified for the defects of interest. The interpretation of inspection results and the assessment of their structural significance should be carried out by suitably qualified and experienced personnel (EMC.30). For crack-like defects, the defect sizes and orientations used in CCCA/DTA analyses should be conservative, and include the contribution associated with the uncertainties in defect location and sizing for the particular inspection technique. The level of conservatism in the analysis, recognising the gap between modelling idealisation and observations of defects, should be appropriate to the overall safety case and consequences of failure.

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

5.89 Examination shortfalls should be clearly identified. The implications of the inability to inspect all areas should be addressed in the assessment of the significance of any defects found or defects that could exist in the inaccessible areas.

Decommissioning

5.90 NS-TAST-GD-026 provides guidance on decommissioning including some factors to be taken into account when plans are made for the 'safe store' or dismantling of graphite reactor cores.

5.91 Stored (Wigner) energy is most likely to be significant in the coolest zones of the oldest reactor cores. Neutron reflectors and other zones operated at relatively low temperatures (such as, in the UK’s production and early research reactors) will be of concern.

5.92 Neutron shields containing graphite and boron may raise safety issues because boron is known to accelerate radiation damage in graphite.

5.93 Before Stage 3 decommissioning (dismantling of plant including that inside a biological shield) [8], graphite samples will normally be taken to establish the important radionuclide content and, if appropriate, whether stored (Wigner) energy is significant and this information will be taken into account in the waste management strategy.

5.94 The key safety functions of a graphite core should be maintained throughout the operating life of a reactor and during decommissioning until it is defuelled. After this, less onerous demands may be appropriate. Changes in the role of graphite cores for the decommissioning phase should be incorporated in the decommissioning safety case. Several of the factors for normal operation of a nuclear facility (*e.g.* in-service inspection, ageing and degradation, materials monitoring) may remain relevant during decommissioning.

6. REFERENCES

[1] *Safety Assessment Principles for Nuclear Facilities, 2014 Edition, Revision 0, ONR, November 2014.*

[2] *Licence Condition Handbook, ONR, February 2017.*

[3] *ONR Guidance NS-ENF-GD-002, The Use of the Enforcement Management Model in ONR, Office for Nuclear Regulation, 2016*

[4] *ONR-ENF-GD-006, Enforcement, Office for Nuclear Regulation, 2018*

[5] *ONR Technical Assessment Guide - NS-TAST-GD-016 - Integrity of Metal Structures, Systems and Components, 2017*

[6] *Heys G B, 'Nuclear Installations Inspectorate Strategy to Secure Adequate and ALARP Graphite Core Safety Cases,' HSE report NSD DIV 1 AR No. 27/05 Issue 1, 2008, TRIM 2007/53348*

[7] *Health and Safety Executive, 'Reducing Risks, Protecting People: HSE's Decision-making Process,' HSE Books, ISBN 0-7176-2151-0, 2001*

[8] *Health and Safety Executive, UKAEA's Strategy for the Decommissioning of Its Nuclear Licensed Sites: A Review by HM Nuclear Installations Inspectorate,' 2002*

[9] *IAEA Safety Standards: Safety of Nuclear Power Plants: Design, Specific Safety Requirements, SSR-2/1 (Rev. 1), 2016*

7. GLOSSARY AND ABBREVIATIONS

ALARP	As low as reasonably practicable
BSL	Basic Safety Level
BSO	Basic Safety Objective
BWR	Boiling water reactor
CCCA	Component and core condition assessment
DSRL	Decommissioning safety reference level
DTA	Defect tolerance assessment
EISST	Examination, inspection, surveillance, sampling and testing
EMM	Enforcement Management Model
HSE	Health and Safety Executive
IAEA	International Atomic Energy Agency
INSA	Independent nuclear safety assessment
ISI	In-service inspection
LC	Licence condition
MTR	Materials test reactor
ONR	Office for Nuclear Regulation
PSA	Probabilistic safety analysis
PSI	Pre-service inspection
PSR	Periodic safety review
PWR	Pressurized water reactor
QA	Quality assurance
R2P2	Reducing Risk Protecting People
SAP	Safety assessment principle
SAR	Safety analysis report
TAG	Technical assessment guide
TECDOCS	Technical documents series (IAEA)
WENRA	Western European Nuclear Regulators Association
WSFSSRL	Waste and spent fuel storage safety reference level

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

A1.1. The WENRA Safety Reference Levels for Existing Reactors were updated in 2014 [A1-1] in relation to lessons learned from TEPCO FUKUSHIMA DAI-ICHI ACCIDENT. They are for existing plants and—from the titles and contents—their application is limited to reactors, predominantly light water reactors.

A1.2. Generally the WENRA Reference Levels are at a high level, somewhat similar to the higher level requirements of the SAPs. A review indicated no significant differences between the issues covered in this TAG and the WENRA Reference Levels. Therefore, if the inspector applies the SAPs, the general WENRA Reactor Safety Reference Levels should be accounted for. A few WENRA Reference Levels are quite specific; relevant examples for graphite reactor cores are set out below.

A1.3. The Reference Levels of most relevance to the assessment of graphite components and structures are Issue I: 'Ageing Management' and Issue K: 'Maintenance, In-service Inspection and Functional Testing'. It is not realistic to give a point-by-point comparison between these WENRA Reference Levels and the SAPs here. Suffice to say that the requirements of the WENRA Reference Levels are captured in this TAG as illustrated in the following examples. Paragraphs 1.1 and 2.1 of Issue I: 'Ageing Management' state:

> *'1.1. The operating organisation shall have an Ageing Management Programme to identify all ageing mechanisms relevant to structures, systems and components (SSCs) important to safety, determine their possible consequences, and determine necessary activities in order to maintain the operability and reliability of these SSCs.'*

which is covered by EAD.2 and associated guidance in this TAG and

> *'2.1 The Licensee shall assess structures, systems and components important to safety taking into account relevant ageing and wear-out mechanisms and potential age related degradations in order to ensure the capability of the plant to perform the necessary safety functions throughout its planned life, under design basis conditions.'*

This is covered by EAD.1 and EAD.2 and associated guidance in this TAG.

A1.4. Paragraph 3.12 of issue K: 'Maintenance, In-service Inspection and Functional Testing' states:

> *'When a detected flaw that exceeds the acceptance criteria is found in a sample, additional examinations shall be performed to investigate the specific problem area in the analysis of additional analogous components (or areas). The extent of further examinations shall be decided with due regard for the nature of the flaw and degree to which it affects the nuclear safety assessments for the plant or component and the potential consequences.'*

This point is not explicitly covered in the SAPs but may be implied by the guidance in §5.37 to §5.53 (component and core condition assessment) and §5.75 to §5.89 (EISST) of this TAG.

A1.5. There are also a number of other Reference Levels which inspectors may find useful in the assessment of safety cases for graphite reactor cores. These are listed below with some relevant points which are considered in this TAG:

Issue B Operating organisation

Issue C	Management system
Issue E	Design basis envelope for existing reactors
Issue F	Design extension of existing reactors
Issue G	Safety classification of structures, systems and components
Issue H	Operational limits and conditions
Issue O	Probabilistic safety analysis
Issue P	Periodic safety reviews
Issue Q	Plant Modifications
Issue T	Natural Hazards

A1.6. Issue B 'Operating Organisation', in particular, (B2.5) emphasises the need for the licensee to ensure that relevant operating experience, international development of safety standards and new knowledge gained through research and development are analysed in a systematic way and continuously used to improve the plant and licensee's activities. Subsection B3 'Sufficiency and competency of staff' includes the availability of SQEP resources in both the licensee's and supporting organizations. Issue C 'Management system' paragraph C4.1, draws attention to the importance of maintaining competent resources to carry out the activities of the licensee. In view of the limited pool of expert knowledge this Reference Level warrants consideration with regard to the long term management of safety for graphite components and structures. Notably, graphite safety cases may adopt novel and/or complex approaches to model materials behaviour.

A1.7. In Issue E 'Design Basis Envelope for Existing Reactors', a key point is the ability to identify ‘states’ relevant to normal operation, transient conditions and accident initiating events. The inspector should seek to be familiar with all probable initiating events which may affect delivery of the safety functions of the core.

A1.8. Issue E paragraphs E9.6 and E9.7 state:

> *'The means for shutting down the reactor shall consist of at least two diverse systems.'*
>
> *'At least one of the two systems shall, on its own, be capable of quickly rendering the nuclear reactor sub critical by an adequate margin from operational states and in design basis accidents, on the assumption of a single failure'*

There may be failures in graphite components and structures which could compromise physical means of shutting down a reactor. This is addressed in this TAG and, dependent on the overall risk, design changes to enhance shutdown systems should be considered as an ALARP improvement.

A1.9. Issue E paragraph E11.1 mentions the importance of undertaking a regular review of the design basis to reflect operating experience and any significant information:

> *'The actual design basis shall regularly, and when relevant as a result of operating experience and significant new safety information, be reviewed, using both a deterministic and a probabilistic approach as well as engineering judgement to determine whether the design basis is still appropriate. Based on the results of these reviews needs and opportunities for improvements shall be identified and relevant measures shall be implemented.'*

Inspectors need to give careful consideration to this Reference Level with developments in the safety cases for graphite components and structures.

A1.10. In Issue F 'Design Extension of Existing Reactors', paragraph F5.1 underlines the importance of undertaking a regular review of design extension conditions.

A1.11. The importance of keeping Operational Limits and Conditions (OLCs) up-to-date and maintaining adequate margins between operational limits and safety limits are emphasised in Issue H paragraph H2.2 and H5:

> *'H2.2 OLCs shall be kept updated and reviewed in the light of experience, developments in science and technology, and every time modifications in the plant or in the safety analysis warrant it, and changed if necessary.*
>
> ***H5. Safety limits, safety systems settings and operational limits***
>
> *H5.1 Adequate margins shall be ensured between operational limits and the established safety systems settings, to avoid undesirably frequent actuation of safety systems.*
>
> *H5.2 Safety limits shall be established using a conservative approach to take uncertainties in the safety analyses into account.'*

A1.12. The role of PSA is discussed in Issue O 'Probabilistic Safety Analysis (PSA)'. In terms of the integrity of graphite components and structures, inspectors should note the requirements of paragraphs O3.2 to O3.4 and O4.1 to O4.3:

> *'O3.2 PSA shall be used to identify the need for modifications to the plant and its procedures, including for severe accident management measures, in order to reduce the risk from the plant.*
>
> *O3.3 PSA shall be used to assess the overall risk from the plant, to demonstrate that a balanced design has been achieved, and to provide confidence that there are no 'cliff-edge effects'.*
>
> *O3.4 PSA shall be used to assess the adequacy of plant modifications, changes to operational limits and conditions and procedures and to assess the significance of operational occurrences.*
>
> *O4.1 The limitations of PSA shall be understood, recognized and taken into account in all its use. The adequacy of a particular PSA application shall always be checked with respect to these limitations.*
>
> *O4.2 When PSA is used, for evaluating or changing the requirements on periodic testing and allowed outage time for a system or a component, all relevant items, including states of systems and components and safety functions they participate in, shall be included in the analysis.*
>
> *O4.3 The operability of components that have been found by PSA to be important to safety shall be ensured and their role shall be recorded in the safety SAR.'*

A1.13. The scope of the PSR is discussed in Issue P Periodic Safety Review (PSR). Paragraph P2.2 identifies a minimum number of areas several of which are relevant to the management of safety cases for graphite components and structures:

> *'P2.2 The scope of the review shall be clearly defined and justified. The scope shall be as comprehensive as reasonably practical with regard to significant safety aspects of an operating plant and, as a minimum the following areas shall be covered by the review:*
>
> - *Plant design*
> - *Actual condition of structures, systems and components (SSCs) important to safety*
> - *Equipment qualification*
> - *Ageing*
> - *Deterministic safety analysis*

- *Probabilistic safety assessment*
- *Hazard analysis*
- *Safety performance*
- *Use of experience from other plants and research findings*
- *Organization, the management system and safety culture*
- *Procedures*
- *Human factors*
- *Emergency planning*
- *Radiological impact on the environment.'*

A1.14. The updated version of WENRA Decommissioning Safety Reference Levels (DSRLs) report was issued in April 2015 [A1-2]. The WENRA DSRLs are generic and do not explicitly contain requirements for graphite components and structures. This TAG covers decommissioning in §5.90 to §5.94. Inspectors should also consult the relevant TAG, NS-TAST-GD-026.

A1.15. The updated version of WENRA Waste and Spent Fuel Storage Safety Reference Levels (WSFSSRLs) report was issued in April 2014 [A1-3]. The WENRA WSFSSRLs do not explicitly contain requirements for graphite components and structures. However, aspects of this TAG may be relevant to assessing the safe storage of graphite waste. Inspectors should also consult the relevant TAG, NS-TAST-GD-024.

REFERENCES

[A1-1] Western European Nuclear Regulators' Association (WENRA), WENRA Safety Reference Levels for Existing Reactors, 2014

[A1-2] WENRA Decommissioning Safety Reference Levels, Version 2.2, 2015

[A1-3] WENRA Waste and Spent Fuel Storage Safety Reference Levels, Version 2.2, 2014

9. APPENDIX A2: IAEA STANDARDS, GUIDANCE AND 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)

A2.2. IAEA Safety Standards comprise Safety Fundamentals, Safety Requirements and Safety Guides; all in the Safety Standards Series. 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.

A2.3. The IAEA documents apply only to nuclear power reactors and mostly to PWR and BWR type reactors. The remit of the ONR Safety Assessment Principles and this TAG is wider and covers all graphite related nuclear facilities. The lists of IAEA documents have been assessed and the subset potentially relevant to assessment of graphite components and structures is set out below.

A2.4. In addition to the search of IAEA documents with titles potentially relevant to assessment of graphite components and structures, a more general search of IAEA documents was made to identify generic IAEA documents relevant to the management of ageing and degradation. A number of documents were identified. These are not primarily concerned with graphite structural integrity; they give an overview of the approaches to the management of ageing and degradation. The documents found as a result of this search and filtering are listed below.

Safety Standard Series

A2.5. SF-1, Safety Fundamentals [A2-1] is the primary publication in the IAEA Safety Standards Series. It is a high level document containing 10 Principles which have equivalents in the SAPs. There are no SF-1 Principles which relate specifically to the assessment of graphite components and cores.

A2.6. Two Requirements documents were identified as potentially relevant to this TAG. These cover design SSR-2/1 [A2-2] and operations NS-R-2 [A2-3]. According to SSR-2/1, IAEA Safety Requirements establish the requirements that must be met to ensure safety. These requirements, which are expressed as 'shall' statements, are governed by the objectives and principles presented in the Safety Fundamentals.

A2.7. Ten Safety Guides have been identified as potentially relevant to this TAG. Although these are aimed at the designer and operator of a nuclear power plant—not the regulator—the inspector may find their contents useful as background information:

- GSG-4 (Use of external experts by the regulatory body) [A2-4]
- NS-G-1.6 (Seismic design and qualification for nuclear power plants) [A2-5]
- NS-G-1.12 (Design of the reactor core for nuclear power plants) [A2-6]
- NS-G-2.2 (Operational limits and conditions and operating procedures) [A2-7]
- NS-G-2.3 (Modifications) [A2-8]
- NS-G-2.6 (Maintenance, surveillance and in-service inspection) [A2-9]
- NS-G-2.12 (Ageing Management for Nuclear Power Plants) [A2-10]
- NS-G-2.13 (Evaluation of Seismic Safety for Existing Nuclear Installations) [A2-11]
- SSG-25 (Periodic safety review for nuclear power plants) [A2-12]

- WS-G-2.1(Decommissioning) [A2-13]

These Guides recommend actions, conditions or procedures to meet safety requirements. The recommendations are expressed as 'should' statements, with the implication that it is necessary to follow them or equivalent alternatives to comply with the requirements.

A2.8. NS-G-2.3 covers modifications. The guide is general; there is nothing in it which is only relevant to graphite reactor cores.

A2.9. NS-G-2.6 covers maintenance, surveillance and in-service inspection. For assessment of integrity of graphite components and structures it is mainly the surveillance and in-service inspection aspects which are relevant. The few paragraphs on plant ageing and plants designed to earlier standards only provide a set of general points. SAPs (2014 Edition) which address surveillance and in-service inspection for graphite components and structures are EAD.1 to EAD.5, EGR.4 and EGR.15. General matters related to maintenance, inspection and testing are covered in SAPs EMT.1 to EMT.8.

Safety Report Series

A2.10. The IAEA Safety Report series provides documents on management related to ageing [A2-14].

Technical Reports and TECDOC Series

A2.11. Documents in the IAEA Technical Report Series and the TECDOC Series provide information on graphite-specific topics, such as [A2-15 to A2-19] and ageing management, such as [A2-20 to A2-23]; these do not set out IAEA Principles, Requirements or Guidance.

REFERENCES

Safety fundamentals

[A2-1] IAEA safety standards: Safety fundamentals SF-1, 'Fundamental Safety Principles: Safety Fundamentals,' 2006

[A2-2] IAEA standards series: Specific safety requirements SSR-2/1 (Rev. 1), 'Safety of Nuclear Power Plants: Design,' 2016

[A2-3] IAEA standards series: Specific safety requirements SSR-2/2 (Rev. 1), 'Safety of Nuclear Power Plants: Commissioning and Operation,' 2016

[A2-4] IAEA Standards Series: General Safety Guide, GSG-4, 'Use of external experts by the regulatory body,' 2013

[A2-5] IAEA Standards Series: Safety Guide, NS-G-1.6, 'Seismic design and qualification for nuclear power plant,' 2003

[A2-6] IAEA Safety standards series: Safety guide NS-G-1.12, 'Design of the reactor core for nuclear power plants,' 2005

[A2-7] IAEA Standards Series: Safety Guide: NS-G-2.2, 'Operational limits and conditions and operating procedures for nuclear power plants: Safety guide,' 2000

[A2-8] IAEA Standards Series: Safety Guide NS-G-2.3, 'Modifications to nuclear power plants: Safety guide,' 2001

[A2-9] IAEA Standards Series: Safety Guide, NS-G-2.6, 'Maintenance, surveillance and in-service inspection in nuclear power plants,' 2002

[A2-10] IAEA Standards Series: Safety Guide, NS-G-2.12, 'Ageing Management for Nuclear Power Plants,' 2009

[A2-11] IAEA Standards Series: Safety Guide, NS-G-2.13, 'Evaluation of Seismic Safety for Existing Nuclear Installations,' 2009

[A2-12] IAEA Standards Series: Specific Safety Guide, SSG-25, 'Periodic safety review for nuclear power plants,' 2013

[A2-13] IAEA Standards Series: Safety Guide WS-G-2.1, 'Decommissioning of nuclear power plants and research reactors: Safety guide,' 1999 (under revision)

[A2-14] International Atomic Energy Agency, Safety Report Series No. 15, 'Implementation and review of a nuclear power plant aging management programme,' 1999

Graphite-specific

[A2-15] IAEA, IAEA-TECDOC-1154, 'Irradiation damage in graphite due to fast neutrons in fission and fusion systems,' 2000

[A2-16] IAEA, IAEA-TECDOC-690, 'The status of graphite development for gas-cooled reactors,' 1993

[A2-17] IAEA, IAEA-TECDOC-1521, 'Characterization, treatment and conditioning of radioactive graphite from decommissioning of nuclear reactors,' 2006

[A2-18] IAEA, IAEA-TECDOC-901, 'Graphite moderator lifecycle behaviour, Proceedings of a Specialists Meeting held in Bath UK,' 1995

[A2-19] IAEA, IAEA-TECDOC-1790, 'Processing of Irradiated Graphite to Meet Acceptance Criteria for Waste Disposal,' 2016

Ageing management

[A2-20] IAEA, TR 338, 'Methodology for the management of ageing of nuclear power plant components important to safety,' 1992

[A2-21] IAEA, IAEA-GNPPA-CD/1, ISBN 92-0-136502-0, CD-ROM Version 1, 'Guidance on ageing management for nuclear power plants,' 2002

[A2-22] IAEA, IAEA-TECDOC—1399, ' Nuclear power industries ageing workforce: Transfer of knowledge to the next generation,' 2004

[A2-23] IAEA, IAEA-TECDOC-540, 'Safety Aspects of Nuclear Power Plant Ageing,' 1990

10. APPENDIX A3: CITATIONS OF GRAPHITE CORE SAPS IN §5 OF THIS TAG

SAP			This TAG	
Safety case area	**Subject**	**Identity**	**Section**	**Subject**
General	Safety cases	EGR.1	5.5, 5.18	Assessment of graphite core safety cases in a broad context
			5.19	Comparison between this TAG and the TAG for metal components and structures
			5.26	Multi-legged graphite core safety cases
Design	Demonstration of tolerance	EGR.2	5.31	Tolerance of the core to ageing and imposed loads
	Monitoring	EGR.3	5.32	Adequacy of design to enable monitoring
	Inspection and surveillance	EGR.4	5.32	Provision of feature for inspection and for samples to monitor in-reactor behaviour
Manufacture, construction and commissioning	Manufacturing records	EGR.5	5.36	Adequacy of case histories
	Location records	EGR.6	5.36	Adequacy of case histories
Component and core condition assessment (CCCA)	Materials properties	EGR.7	5.37 & 5.38	Predictions of graphite component properties and condition
	Predictive models	EGR.8	5.44	Caution in assessing biased models
			5.45	Validation of models
	Materials properties data	EGR.9	5.47	Extrapolation and interpolation from data
Defect tolerance assessment (DTA)	Effects of defects	EGR.10	5.54	Objective of the DTA
			5.56	Action if a DTA is unable to demonstrate that safety functionality is achieved
	Safe working life	EGR.11	5.57	Margin between the operating and fault envelope
			5.59	Maintenance of safe operating envelope
	Margins	EGR.12	5.59	Maintenance of safe operating envelope
	Use of data	EGR.13	5.50	Use of conservative data and sensitivity studies
Monitoring	Monitoring systems	EGR.14	5.74	Sufficiency of monitoring
Examination, inspection, surveillance, sampling and testing (EISST)	Extent and frequency	EGR.15	5.75	Adequacy of EISST