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Material Compatibility for non-Light Water Reactors

Interim Staff Guidance

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INTERIM STAFF GUIDANCE

MATERIAL COMPATIBILITY FOR NON-LIGHT WATER REACTORS

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PURPOSE

This document provides interim staff guidance (ISG) to assist the U.S. Nuclear Regulatory Commission (NRC) staff in reviewing applications for construction and operation of non-light water reactor (non-LWR) designs, including power and non-power reactors. The guidance in this document identifies areas of staff review that could be necessary for a submittal seeking to use materials allowed under American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME Code), Section III, "Rules for the Construction of Nuclear Facility Components," Division 5, "High Temperature Reactors" (Section III-5) (ASME, 2017). Section III-5 specifies the mechanical properties and allowable stresses to be used for design of components in high-temperature reactors (HTRs). However, as stated in Section III-5, HBB-1110(g), the ASME Code rules do not provide methods to evaluate deterioration that may occur in service as a result of corrosion, mass transfer phenomena, radiation effects, or other material instabilities. This ISG identifies information that the staff should consider as part of its evaluation of a non-LWR application to review applicable design requirements including environmental compatibility, qualification, and monitoring programs for safety-related, safety-significant, and, as needed, non-safety-related structures, systems, and components (SSCs). The actual information necessary for reviewing qualification and monitoring programs would depend on many factors, such as plant design, importance to safety of structures, systems, and components, specific environments, and maturity of research in a given area. The staff should consider these concepts for non-LWR applications for construction permits or operating licenses under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," and non-LWR applications for design certifications, combined licenses, standard design approvals, or manufacturing licenses under 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants."

BACKGROUND

In its review of non-LWR applications, the NRC evaluates whether structural materials will allow components to fulfill design requirements for the design life, or that adequate surveillance and monitoring programs are in place. Regulations in 10 CFR Part 50 and 10 CFR Part 52 include requirements for material qualification and performance monitoring. The staff identified the need for guidance on appropriate qualification, performance monitoring methods, and in-service inspection to support the staff reviews of applications for a construction permit or operating license under 10 CFR Part 50 or for a design certification, combined license, standard design approval, or manufacturing license under 10 CFR Part 52 that proposes to use materials allowed under ASME Section III, Division 5.

New fabrication methods present different material considerations for staff reviews. As an alternative to conventional manufacturing processes (e.g., forging, castings), an applicant may propose components fabricated with advanced manufacturing technologies (AMTs), such as laser powder bed fusion or directed energy deposition additive manufacturing. These techniques can produce materials with different microstructures or types of defects than those of conventional metal manufacturing. Postprocessing requirements may also differ. Therefore, it is important that appropriate controls on manufacturing be applied to ensure that components with acceptable properties are manufactured and that proper testing is conducted to confirm material properties. The information related to AMTs that the staff would need to review depends on many factors, including the maturity of the AMT process in codes and standards, applicable precedents, as well as the safety and risk significance of the intended use of the component. The NRC is in the process of developing both generic (NRC, 2021a) and AMT-specific guidelines (e.g., NRC, 2021b) for considering the following elements of a submittal that may use AMT components: quality assurance (QA), AMT process qualification, supplemental qualification testing, production process control and verification, and performance monitoring.

Non-LWRs present operational environmental challenges to material performance due to differences in operating temperatures and types of coolants from currently operating light water reactors (LWRs). Operating temperatures of non-LWRs may be significantly higher than those currently used in nuclear power plants. Non-LWRs may operate in temperature ranges corresponding to the creep regime in which deformation may occur with applied stress. The NRC developed Regulatory Guide (RG) 1.87, Revision 2, "Acceptability of ASME Code, Section III, Division 5, 'High Temperature Reactors,'" issued January 2023 (NRC, 2023a; NRC, 2023b; NRC, 2022a), which endorses the use of Section III-5, with conditions. Section III-5 considers mechanical and thermal stresses due to cyclic operation and high-temperature creep in air; however, it does not cover degradation that may occur in service as a result of radiation effects, corrosion, erosion, thermal embrittlement, or instability of the material.[1] Another consideration is that the coolants used in non-LWRs are significantly different from those used in LWRs. These coolants may be liquid metals (e.g., sodium, lead), liquid salts with or without fuel, helium, or possibly other coolants not yet considered. These different coolant environments may increase susceptibility to material corrosion, degradation mechanisms, and irradiation effects. Studies have identified the gaps in knowledge that exist for some of these coolant types and the impact on the materials being considered in the construction and operation of these non-LWR nuclear power plants (NRC, 2003; INL, 2006; ANL, 2017; ORNL, 2019; NRC, 2021c; NRC, 2021d; NRC, 2021e; EPRI, 2019a; EPRI, 2019b; EPRI, 2020a; EPRI, 2020b). This ISG provides NRC staff guidance in reviewing materials areas that are not covered by ASME Section III, Division 5. The ISG identifies information the staff should consider in its review related to materials qualification. It also indicates where monitoring and surveillance may be appropriate to be relied upon to ensure component integrity.

As noted above, the variety of coolants proposed for non-LWR designs create unique operating environments for reactor materials and components. This ISG provides non-plant-specific guidance for non-LWRs in the discussion section below. In addition, Parts 1, 2, and 3 of this ISG provide technology-specific guidance for molten salt reactors (MSRs), liquid metal reactors, and high-temperature gas-cooled reactors, respectively.

APPLICABILITY

This ISG is applicable to NRC staff reviews of applications for non-LWR designs, including power and non-power reactors, for permits, licenses, certifications, and approvals under 10 CFR Parts 50 and 52. As stated in the Commission's Policy Statement on the Regulation of Advanced Reactors (73 FR 60612; October 14, 2008), advanced designs are expected to

[1] ASME Code, Section III, Division 5, paragraph HAA-1130, "Limits of These Rules"

provide enhanced margins of safety; use simplified, inherent, passive, or other innovative means to accomplish their safety and security functions; or both.

Qualification of non-Code materials is outside the scope of this ISG. However, if an applicant adequately qualifies a material to Section III Division 5 rules, the staff should ensure the considerations in this ISG are addressed when reviewing compatibility of these materials with the respective environments.

GUIDANCE

Current Regulatory Framework

Under 10 CFR 50.34(a)(3)(i), 10 CFR 52.47(a)(3)(i), and 10 CFR 52.79a(4)(i), applicants must include principal design criteria (PDC) for the facility.

Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 applies to LWRs but is also considered to be generally applicable to other types of nuclear power units and is intended to provide guidance in establishing the principal design criteria for such units.

For non-LWRs, RG 1.232, "Guidance for Developing Principal Design Criteria for Non-Light Water Reactors," issued March 2018 (NRC, 2018), provides proposed guidance for the development of PDCs for non-LWR designs. The RG also describes the NRC's proposed guidance for modifying and supplementing the general design criteria to develop PDC that address two specific non-LWR design concepts: sodium-cooled fast reactors and modular high-temperature gas-cooled reactors. The following criteria are related to material qualification for structural materials:

- Advanced Reactor Design Criterion (ARDC) 4, Sodium Fast Reactor Design Criterion (SFR-DC) 4, and Modular High Temperature Gas-Cooled Reactor Design Criterion (MHTGR-DC) 4 states, in part, that SSCs important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents.

- ARDC 14 states that the reactor coolant boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

- ARDC 30, and MHTGR-DC 30 states, in part, that components that are part of the reactor coolant boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical.

- ARDC 31 states, in part, that when stressed under operating, maintenance, testing, and postulated accident conditions, (1) the reactor coolant boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized.

- ARDC 32 states that components that are part of the reactor coolant boundary be designed to permit periodic inspection and functional testing of important areas and features to assess their structural and leak-tight integrity and have an appropriate material surveillance program for the reactor vessel.

- SFR-DC 71 states, in part, that necessary systems shall be provided to maintain the purity of primary coolant sodium and cover gas within specified design limits.

- SFR-DC 74 states, in part, that SSCs containing sodium shall be designed and located to avoid contact between sodium and water and to limit the adverse effects of chemical reactions between sodium and water on the capability of any SSC to perform any of its intended safety functions.

- SFR-DC 75 states that components that are part of the intermediate coolant boundary shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed.

- SFR-DC 76 states that the intermediate coolant boundary shall be designed with sufficient margin so that, when stressed under operating, maintenance, testing, and postulated accident conditions, (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized.

- SFR-DC 77 states that components that are part of the intermediate coolant boundary shall be designed to permit (1) periodic inspections and functional testing of important areas and features to assess their structural and leak-tight integrity commensurate with the system's importance to safety and (2) an appropriate material surveillance program for the intermediate coolant boundary.

Although RG 1.232 does not contain design criteria specifically for MSRs, many of the criteria in the ARDC and some SFR-DC will likely apply to MSRs. Additionally, an applicant using an MSR design may propose additional design criteria not discussed in RG 1.232. Additionally, while RG 1.232 does not explicitly consider non-LWR non-power reactors, the design criteria listed above may be used to inform the development of PDC related to material qualification for structural materials at non-LWR non-power reactors.

The staff should confirm that sufficient information with regards to materials qualification, mitigation strategies, performance monitoring, and surveillance programs is provided to demonstrate established facility specific PDCs are satisfied.

Discussion

Qualification and Performance Monitoring

This ISG identifies information that the staff should consider during its review of applications using ASME Section III, Division 5 qualified materials. An SSC's performance should be demonstrated through a combination of materials qualification programs, supplemental testing, and performance monitoring and surveillance programs, which collectively provide assurance that a component will meet the design requirements over its intended design life in the applicable operating environment.

Quality assurance (QA) is a process followed to ensure that a component adheres to quality requirements (e.g., a program meeting the criteria in Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50). Attributes of a QA program include procedures, recordkeeping, inspections, corrective actions, and audits. QA programs establish requirements for process qualification and production process control, and possibly also establish requirements for supplemental testing, performance monitoring, and surveillance programs. The staff should confirm that an appropriate QA program was followed when reviewing a materials qualification program[2].

The selection of structural materials for the reactor design should consider effects on the materials properties and allowable stresses due to interactions with the operating environment. Materials qualification and monitoring programs should include testing conducted, or use of historical data collected, in an environment simulating the anticipated operating environment for the reactor, including chemical environment, temperatures, and irradiation. It is incumbent upon the applicant to demonstrate that data is directly applicable to the plant design and environment. Testing or historical data should account for uncertainties in the environment, material composition, fabrication methods, and operating conditions. The scope of this testing should include safety-related component materials, safety-significant component materials, and, as needed, non-safety related component materials whose failure could impact critical design functions. Testing should be conducted to determine if materials properties and allowable stresses meet applicable codes and standards or other design requirements. If necessary, appropriate reduction factors should be applied to the materials properties and allowable stresses from the applicable design codes and/or design specifications.

Performance monitoring and surveillance programs are used in tandem to ensure that the component will continue to meet its design requirements until the end of its intended design life. While performance monitoring typically consists of inspections or examinations to confirm adequate performance and to identify unacceptable degradation, it may also include aging management programs or post-service evaluations. Examples of this type of performance monitoring that could be appropriate include chemistry, temperature, or flow monitoring, as well as wall thickness measurements. Surveillance programs include examination of test coupons and components removed from the reactor over the licensed operating period. Data gathered from surveillance programs provides physical data which is then used to help construct and benchmark models for predicting the degradation of components within the reactor. For components for which there is little data on performance in similar operating environments and conditions, performance monitoring and surveillance programs could be an acceptable way to show that the component will maintain its intended function throughout the design life. A component with a significant design margin or one that has demonstrated acceptable performance under similar operating environments and conditions may require less rigorous performance monitoring and surveillance programs. The staff review should include performance monitoring and surveillance programs for SSCs that are not planned to undergo periodic inspections and/or functional testing.

Qualification and performance monitoring should be targeted to provide a holistic aging management strategy over the intended design life of the components. ASME Section XI-2 provides one method for developing a comprehensive aging management strategy, subject to NRC acceptance of the proposed program. The NRC endorsed ASME Section XI-2, subject to certain conditions, for use by non-LWR applicants and licensees in RG 1.246, "Acceptability of ASME Code, Section XI, Division 2, 'Requirements for Reliability and Integrity Management (RIM) Programs for Nuclear Power Plants,' for Non-Light Water

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[2] While a quality assurance program description is not required to be submitted or approved as part of a non-power reactor operating license application, as part of its review of an application, the staff will determine whether a non-power reactor applicant considered how to appropriately qualify materials to support the design and licensing of facilities as part of the development of managerial and administrative controls to be used to assure safe operation, as required by 10 CFR 50.34(b)(6)(ii).

Reactors," issued October 2022 (NRC, 2022b). ASME Section XI-2 requires an applicant to develop strategies for inspection, monitoring, and repairing SSCs throughout the design lifetime. Although RG 1.246 proposes one method the NRC finds acceptable, applicants may propose other methods.

General Degradation Mechanisms

Below are degradation mechanisms that are likely to apply across different reactor designs, operating environments, and materials. The degradation mechanisms identified reflect the current state of knowledge. As additional operating experience and laboratory testing become available, the way in which each identified degradation mechanism should be addressed may change and new degradation mechanisms may be identified. In the meantime, the staff should evaluate whether applicants have adequately addressed the following general degradation mechanisms for various reactor environments.

Carburization

Formation of chromium carbides promotes carburization of structural alloys which can increase degradation rates of these materials (Chan 2018, NRC, 2003, NRC 2021d, NRC 2021e Sridharan, 2019). As noted in the design specific appendices below, there are different causes for carburization or decarburization depending on the environment and other materials present in the design. The staff should review interactions between graphite and carbon impurities in the coolant with metals to ensure that qualification, monitoring, surveillance, or inspection programs address potential carburization.

Corrosion

The staff should ensure that corrosion is assessed as a function of temperature; time; microstructure; coolant composition and chemistry; and coolant flow conditions, including, as appropriate, synergistic effects of irradiation. Additionally, localized corrosion, galvanic effects, leaching, erosion/wear, and coolant solubility-driven corrosion effects should be considered. The staff should confirm that applicants also considered appropriate mitigation strategies, performance monitoring, and surveillance programs to ensure that SSCs affected by corrosion continue to satisfy the design criteria for the facility.

Creep and Creep-Fatigue

The staff should ensure that changes to the materials properties and allowable stresses of ASME Code Section III-5, or other applicable design codes, are assessed as a function of irradiation time, temperature, and environment. Affected properties include the time-dependent allowable stress (S_t), rupture stress (S_r), creep-fatigue diagram, fatigue curves, and isochronous stress-strain curves. The staff should verify that applicants also consider appropriate mitigation strategies, performance monitoring, and surveillance programs to ensure that SSCs affected by creep-induced degradation mechanisms continue to satisfy the design criteria.

Environmentally Assisted Cracking

The staff should ensure that environmentally assisted cracking mechanisms are assessed, including stress corrosion cracking (SCC), intergranular cracking (IGC), and fatigue cracking. Based on operating experience and laboratory studies conducted in LWRs, it is expected that environmentally assisted cracking is most likely to be significant in weld metal or in the heat-affected zone. It is important that component design minimizes the potential for crack initiation and that that there is sufficient flaw tolerance to fabrication and service cracking. The staff should verify that applicants also consider appropriate mitigation strategies, performance monitoring, and surveillance programs to ensure that SSCs affected by environmentally assisted cracking continue to satisfy the design criteria.

Flow-Induced Degradation (e.g., Abrasion, Erosion, Cavitation)

The staff should ensure that abrasion and erosion of SSCs in contact with the coolant are assessed as a function of temperature, time, microstructure, coolant composition and, as appropriate, chemistry and coolant flow conditions. In addition to potentially undergoing activation, thus contributing to the coolant's activation level, erosion products from SSCs have the potential for depositing elsewhere in the coolant flow path, affecting coolant flow patterns and local heat transfer properties. Additionally, staff should ensure pumps are qualified and tested under operating conditions and coolant flow paths and flow rates are evaluated to minimize the potential for cavitation. The staff should confirm that applicants also consider appropriate mitigation strategies, performance monitoring, and surveillance programs to ensure that SSCs affected by abrasion and erosion continue to satisfy the design criteria.

Flow-Induced Vibration

The staff should evaluate the effects of coolant flow-induced vibrations, which may cause fretting and fretting-assisted fatigue. In addition, the staff should confirm that the flow-induced excitations do not have a frequency close to the natural frequency of the system. The staff should confirm that applicants also considered appropriate mitigation, performance monitoring, and surveillance programs to ensure that SSCs affected by flow-induced vibration continue to satisfy the design criteria.

Irradiation

The staff should evaluate data on the effects of neutron irradiation on materials, including mechanisms such as irradiation-assisted creep, irradiation embrittlement, irradiation assisted SCC, helium embrittlement (Briggs, 1969), and decreased resistance to oxidation. The staff should also evaluate the potential for irradiation-induced swelling in alloys, particularly for alloys containing appreciable amounts of nickel. Asymmetrical irradiation can potentially change component dimensions or mechanical properties such that they no longer meet their design function(s). As such, irradiation effects on such components must be considered. In addition, the staff should consider how activation and fission products in the coolant may accelerate or introduce new irradiation-assisted degradation mechanisms. The staff should verify that applicants also consider appropriate performance monitoring and surveillance programs to ensure that SSCs affected by irradiation continue to satisfy the design criteria. Test specimens within the reactor that can be withdrawn (e.g., coupon specimens irradiated during reactor operations) and tested throughout the operating phase of the reactor could be an appropriate supplement to a materials qualification program; for example, to support longer lifetimes or supplement areas with minimal existing data.

Guidance related to the irradiation and oxidation of graphite is provided in RG 1.87, Rev 2 "Acceptability of ASME Code, Section III, Division 5, 'High Temperature Reactors,'" issued January 2023 (NRC, 2023a), which endorses ASME Code Section III, Division 5, subject to limitations and conditions. NUREG-2245, "Technical Review of the 2017 Edition of ASME Code, Section III, Division 5, 'High Temperature Reactors,'" issued January 2023 (NRC, 2023b), contains the technical basis for RG 1.87. It should be noted that, in general, irradiation induced changes to graphite material properties will undergo a reversal (i.e., increasing values will change to decreasing values) with increasing received neutron dose after reaching the "turnaround dose". Turnaround dose is the critical dose level where this change occurs and indicates when the graphite material irradiation induced dimensional volumetric densification reverses to a volumetric expansion behavior. For example, the strength of irradiated graphite will increase gradually as the dose increases up to the turnaround dose and then will rapidly decrease as dose continues to increase. The exception to this irradiated material behavior is thermal diffusivity, which experiences an immediate and significant decrease in value followed by a gradual decrease in value with increasing accumulated dose after reaching the turnaround dose.

Stress Relaxation Cracking

The staff should ensure that the potential for stress relaxation cracking (SRC) is assessed. As per RG 1.87, applicants should submit a plan for addressing SRC. Also called "reheat cracking," SRC is a mechanism that causes accelerated creep cracking in the weld heat-affected zone due to relaxation of residual stresses. It can lead to premature failure of components in high-temperature service. Several factors, including, but not limited to, weld residual stresses, cold work, larger grain sizes, multiaxial stresses, notches, and constraints caused by the weld joint design, promote SRC. SRC occurs in austenitic alloys within specific temperature ranges characteristic for each individual alloy (Colwell and Shargay, 2020; Shoemaker et al., 2007; van Wortel, 2007; Miller, 1998; API, 2017; NRC, 2019; ASME, 2020; ASME, 2021). Factors to reduce susceptibility include heat treatments, control of alloy composition, control of grain size, and controls on welding (Colwell and Shargay, 2020; van Wortel, 2007; Shoemaker et al., 2007). The staff should confirm that applicants consider appropriate preventive measures during design, construction, and operation, such as in the event of post-startup weld repairs.

Thermal Aging

The staff should evaluate whether the application adequately addresses the effects of thermal aging on metallic components over the design life of the reactor. Microstructural changes as a result of thermal aging are known to result in changes to the mechanical properties of metallic alloys—specifically, a decrease in ductility and fracture toughness. Thermal aging may also result in a decrease of corrosion resistance due to the formation of metallic carbides involving elements expected to form protective oxide layers.

The staff should verify that applicants consider appropriate mitigation strategies, performance monitoring, and surveillance programs to ensure that SSCs affected by thermal aging continue to satisfy the design criteria. If surveillance testing coupons are to be used to measure the effect of thermal aging on the mechanical properties of metallic components, the conditions chosen should be the most conservative, which may not necessarily be at the highest operating temperatures.

Thermal Emissivity

Emissivity is important in calculating heat transfer during operation and accident scenarios, and generally, higher emissivity is desired to assist in radiating heat (NRC, 2021c). Surface roughness can affect emissivity. In addition, the thermally grown surface oxide or carbide can affect emissivity.

The staff should confirm that applicants have considered the impact of exposure to the coolant or ambient air at elevated temperatures on the emissivity of materials if the reactor design specifications rely on thermal emissivity (e.g., for heat rejection). Considerations should include changes to emissivity due to prolonged exposure during normal operating conditions and changes induced under accident conditions.

Thermal Fatigue and Transients

The staff should evaluate whether an application adequately addresses thermal fatigue and transients. These include: (1) the effects of startup testing, which may introduce additional thermal fatigue damage for which the plant was not designed; (2) the potential for thermal striping and thermal stratification, which may occur when coolant streams at different temperatures mix in the vicinity of a component (e.g., a heat exchanger or nozzle); and (3) load following, which may increase the potential for thermal fatigue. To minimize the potential for thermal striping or stratification, the staff should ensure that the application addresses the system design and operational criteria for components with the potential of thermal expansion mismatch caused by the mixing of coolant flows at different temperatures. The staff should ensure that very high cycle fatigue due to thermal striping has been adequately addressed by the applicant.

The staff should also consider potential thermal transients (including startup and shutdown) and the impacts on the reactor that are not addressed through ASME Code design rules. For example, operational experience has shown that thermal transients in HTRs can loosen shrink-fit components. The staff should confirm that applicants also consider appropriate mitigation strategies, performance monitoring, and surveillance programs to ensure that SSCs affected by thermal fatigue and transients continue to satisfy the design criteria.

The staff should verify that, whenever applicable, synergistic effects of thermal fatigue, vibratory fatigue, and creep-fatigue are addressed by the applicant.

Coolant Flow, Wear, and Fretting

The staff should consider the potential impacts of the specific coolant environment on wear and fretting, particularly in heat exchangers and steam generators. Reactor operation may be affected by tribological effects such as friction, wear and fretting. Depending on the reactor design, the interaction between the coolants (as a result of wear and fretting) in the primary, secondary, and steam-generating loops may have adverse consequences for the reactor. For example, fretting of steam generator tubing in sodium fast reactors has historically caused tube leaks that resulted in highly exothermic sodium-water reactions.

Due to the soft nature of graphite and composite core components, the coolant flow as well as any entrained particles in the coolant may induce wear. Important factors for the staff to consider during its review include the coolant density, coolant velocity, and whether dust or small particulates from previous wear could be present.

General Materials Issues

Below are materials topics that are likely to apply to a variety of reactor designs, coolants, and materials. The issues identified reflect the current state of knowledge. As additional operating experience and laboratory testing become available, the way in which each identified issue should be addressed may change and new issues may be identified. The staff should evaluate whether applicants have adequately addressed the following design neutral materials issues as appropriate for their specific application and design.

Advanced Manufacturing Technologies

The staff should evaluate whether an application containing AMT components considers (1) the differences between the AMT and traditional manufacturing methods; (2) the safety significance of the identified differences; (3) the aspects of each AMT that are not currently addressed by codes and standards or regulations; and (4) the impacts of the proposed reactor type, operating conditions, and material on the AMT qualification and performance. It is particularly important that an application fully addresses AMT material performance at high temperatures. Limited studies have shown long-term creep, fatigue, and creep-fatigue properties may be reduced compared to wrought material values (INL 2020, INL 2021). The staff should confirm that applicants also consider appropriate mitigation strategies, performance monitoring, and surveillance programs to ensure that SSCs fabricated by AMTs continue to satisfy the design criteria.

Metallic Materials Qualification

The staff should verify that metallic materials to be used in structural components in all reactor designs have been qualified for use in a representative environment. Specifically, the metallic materials should be tested under conditions representative of the anticipated operating environment in terms of temperature, impurity levels, and the potential for oxidation, carburization, decarburization, and other degradation mechanisms, as appropriate, resulting from the reactor environment. The staff should review metallic cladding (ORNL, 2021) to ensure it is qualified in a representative environment with additional considerations given to adherence to their metallic substrate and galvanic coupling. The staff should confirm that applicants also consider appropriate mitigation strategies, performance monitoring, and surveillance programs to ensure that metallic materials and coatings continue to satisfy the design criteria.

Ceramic Insulation

The staff should evaluate whether an application adequately addresses environmental effects of ceramic insulation. The staff should confirm an application considers chemical compatibility of ceramic insulation with the coolant (Sauvage, 1979) and the potential for off-gassing from ceramic insulation. The staff should be aware that off-gassing may affect the performance of sensors located near the insulation during operational, anticipated operational occurrences and accident conditions (Guidez, J. and Prèle, G., 2017).

Dissimilar Metal Welds

Section III-5 provides stress rupture factors to account for the reduced creep strength of welds for the five materials approved for use in Class A, high-temperature components, but these factors do not generally apply to dissimilar metal welds (DMWs), such as welds between ferritic low-alloy steels and austenitic alloys. These bimetallic welds may have creep lifetimes less than those of either the ferritic low-alloy steel or austenitic alloy (EPRI, 2020a). Different coefficients of thermal expansion for the weld constituents and high-temperature solid-state diffusion driven compositional gradients in different alloys are two examples of metallurgical phenomena that can contribute to the reduced lifetime of DMWs. Therefore, the staff should evaluate whether the potential lower lifetimes of DMWs, particularly between ferritic low-alloy steels and austenitic alloys, have been adequately addressed. The staff should verify that applicants have also considered appropriate mitigation strategies, performance monitoring, and surveillance programs to ensure that DMWs continue to satisfy the design criteria.

Monolithic Silicon Carbide, Carbon-Carbon Composites, and Silicon Carbide Composites

The thermomechanical properties, irradiation behavior, and corrosion resistance of monolithic silicon carbide (SiC), carbon-carbon composites (C/C) and silicon carbide composites (SiC/SiC), will depend on the manufacturing method, porosity, and chemical purity (ORNL, 1995; Snead, 2007; ORNL, 2018).

The staff should be aware that nonmetallic composites have the potential for use in non-LWR designs. The 2021 edition of ASME Section III, Division 5, provides a qualification program for nonmetallic composites, which the staff should consider in the review of these materials; however, the staff has not reviewed or endorsed this portion of the Code at the time of writing this ISG (ASME, 2021).

The variability of properties of SiC/SiC will include all the processing parameters affecting monolithic SiC for the constituent parts of the composite, e.g., the fibers, matrix, and fiber/matrix interface in addition to synergistic effects between the constituent parts of the composite.

The NRC staff should review the compatibility of composites with the coolant environment based on the factors discussed above. The staff should confirm that applicants consider appropriate monitoring, and surveillance programs to ensure that SSCs fabricated with these composites continue to satisfy the design criteria.

Gaskets and Seals

The staff should verify when reviewing the application that all gaskets and seals are chemically compatible with the coolant and consider the consequences of corrosion products from the gaskets and seals entering the coolant as well as the consequences of gasket/seal failure on the reactor operation. The staff should also verify that applicants consider appropriate mitigation, performance monitoring, and surveillance programs to ensure that gaskets/seals in contact with coolant continue to satisfy the design criteria.

Reactor-Specific Guidance, Part 1: Molten Salt Reactors

Below are additional degradation considerations likely to apply to MSRs that the staff should consider in its review. MSR designs fall into two categories: liquid fuel and solid fuel. In a liquid-fuel MSR, the fissile material is directly dissolved in the coolant. In a solid-fuel MSR, the fissile material and fission products are typically contained within a TRISO (tristructural isotropic particle fuel) fuel particle, which could be in a prismatic graphite compact or pyrolytic graphite sphere. Additionally, relatively small quantities of fission products may be present in the molten salt coolant. MSRs can use a fast neutron or thermal neutron spectrum. Both types of MSR designs operate at near ambient pressures. Molten salt is generally corrosive to traditional metallic SSCs. Corrosion can be enhanced by galvanic coupling and, in the case of liquid-fuel MSRs, interactions with fissile material and fission products. The Molten Salt Reactor Experiment prototype at Oak Ridge National Laboratory is the only reported example of an operational power MSR (EPRI, 2019a). This section offers details on the design and/or environment specific aspects of the general degradation mechanisms described in the "General Degradation Mechanisms" section above. The staff should evaluate whether applicants have adequately addressed the following materials issues, including plans to monitor, evaluate, and mitigate degradation.

Graphite

Graphite-salt compatibility considerations include fluorination of the graphite and formation of carbides (uranium carbide, chromium carbide, and others), as well as potential infiltration of molten salt into the graphite (NRC, 2021d). The staff should confirm that graphite qualification, monitoring, surveillance, or inspection programs address any potential chemical compatibility issues, as applicable.

Formation of chromium carbides promotes carburization of structural alloys, which can increase degradation rates of these materials (Chan 2018, NRC 2021d). The staff should review interactions between graphite and metals to ensure that qualification, monitoring, surveillance, or inspection programs address potential carburization.

The staff should evaluate whether the application adequately addressed the potential for formation of uranium and other metal carbides on graphite, and subsequent deleterious effects on reactor materials (EPRI, 2019a; NRC, 2021d).

The staff should evaluate whether the application adequately addressed the potential for enhanced corrosion caused by graphite in contact with metallic materials. Increased corrosion of the stainless steel has been observed when graphite and 316L stainless steel are present in the same electrochemical environment (Qiu et al., 2020).

The staff should evaluate whether the application adequately addressed whether the porosity or grain size of the graphite components allows for salt infiltration. If so, the effects of salt intrusion into the graphite should be assessed to determine if this causes any cracking or flaw generation in the graphite, thereby shortening the effective life of the graphite.

The staff should evaluate whether the application adequately addressed the potential for molten salt to accelerate the wear, abrasion, and/or erosion between graphite components. The staff should verify that applicants also considered appropriate mitigation strategies, performance monitoring, and surveillance programs to ensure that SSCs fabricated with graphite continue to satisfy the design criteria.

Materials Considerations

The staff should evaluate whether the application adequately addressed the potential for additional degradation concerns in liquid fueled MSRs when the fissile material is dissolved in the coolant. Fission products will also contribute to the contaminants in the liquid salt and must be considered in the effects on materials wetted by the salt.

The staff should evaluate whether the application adequately addressed the potential for tellurium (Te)-induced cracking in structural alloys and evaluate mitigation strategies, performance monitoring, and surveillance programs to ensure that SSCs satisfy the design criteria. Te has led to IGC of nickel-based alloys (ORNL, 1977; ORNL, 1978). Based on electron probe microanalysis, X-ray diffraction, and transmission electron microscopy (Ignatiev, 2013), Te-induced IGC is likely caused by preferential diffusion of Te along the grain boundaries, followed by formation of the brittle metallic telluride compounds on the grain boundaries and the interface of intergranular carbides.

The staff should evaluate whether the application adequately addressed whether radiation damage to the molten salt could increase its corrosivity due to radiolytic decomposition of the salt over applicable temperature ranges, which may lead to deleterious effects on structural performance. Recombination rates were shown to be fast relative to radiolytic decomposition at high temperatures but not at lower temperatures (ORNL, 1970).

The staff should evaluate whether the application adequately addressed whether corrosion products from structural alloys could affect degradation rates for SiC/SiC composites used as structural components (excluding fuel, as this is not within the scope of this guidance). For example, chromium carbides may be formed by Cr^{3+} from Hastelloy N which may cause accelerated corrosion of SiC (ORNL, 2018).

The staff should confirm that applicants also considered appropriate mitigation strategies, performance monitoring, and surveillance programs to ensure that SSCs in all environments continue to satisfy the design criteria.

Salt Composition

The staff should evaluate whether the application adequately addressed the effects of salt composition on the degradation of metallic and nonmetallic materials due to molten salt, which may lead to deleterious effects on structural performance due to increasing the likelihood of crack initiation or a reduction in strength or ductility. The staff should consider the effects of oxidizing impurities, as well as the impact of reducing agents. Oxidizing impurities include fission products (although these may be limited in a fluoride salt cooled high-temperature reactor design), as well as water and air, and tritium for salts that contain lithium (EPRI, 2019a; NRC, 2021d). Tritium can increase the corrosivity of a lithium-bearing molten salt (NRC, 2021d) by forming tritium fluoride.

The staff should also evaluate whether the application adequately considered the effectiveness of methods to control salt composition and the redox chemistry of the salt (Olander, 2002). These could include the following:

- gas phase control (e.g., HF/H_2)
- major metal control (e.g., Be^{2+}/Be)
- dissolved salt control (e.g., $U^{4+}/U^{3}+$ or Ce^{3+}/Ce^{4+})

The staff should verify that applicants considered appropriate mitigation strategies, performance monitoring, and surveillance programs to ensure that salt composition does not exceed allowable limits that are needed to ensure that component integrity satisfies the design criteria.

Reactor-Specific Guidance, Part 2: Liquid Metal Reactors

Liquid metal reactors are characterized by their operation at or near ambient pressure using a fast neutron spectrum in which the fuel, with metallic cladding, is cooled by liquid sodium, lead, or the lead-bismuth eutectic ((LBE, 44.5 wt% Pb and 55.5 wt% Bi)). The sodium-cooled fast reactor (SFR) has had decades of experience at the experimental, prototype, and commercial scales. The lead fast reactor (LFR) uses liquid lead or LBE as the coolant (EPRI, 2019b) and the design concepts span a range of operating temperatures from 550–800 degrees C. To date, operational experience with LFRs is limited to the development of LBE-cooled reactors for the Alfa-class submarines operated by the Soviet Union from 1967–1983 (EPRI, 2019b; Alemberti, 2014; IRSN, 2012,). More recently, however, construction began on the first prototype lead-cooled reactor, the BREST-OD-300, in 2021 in the Russian Federation (Proctor, 2021). This section offers details on the design and/or operating environment-specific aspects of the general degradation mechanisms described in the “General Degradation Mechanisms” section above.

Sodium Coolant

Below are additional degradation considerations likely to apply to sodium-cooled liquid metal reactors that the staff should consider in its review. The staff should evaluate whether applicants have adequately addressed these considerations. The staff should also ensure that applicants consider appropriate mitigation strategies, performance monitoring, and surveillance programs to address these considerations, such that component integrity satisfies the design criteria.

Caustic Stress-Corrosion Cracking

The staff should evaluate whether the application adequately addressed the potential for caustic SCC, characterized by transgranular and intergranular cracking of a metal in contact with the caustic solution. For example, in the presence of moisture, metallic sodium forms sodium hydroxide, which can induce caustic SCC in some alloys. Certain components, such as steam generators, are more susceptible to the ingress of moisture and therefore to caustic cracking caused by sodium hydroxide (NRC, 2019). Operational experience of the Phoénix reactor demonstrated austenitic stainless steels used in the steam generator were vulnerable to caustic SCC following small leaks and subsequent repairs (Sauvage, 1979). Higher nickel alloys are less susceptible to caustic SCC (Jones, 1992). The staff should verify that designs minimize the potential for interaction of sodium with water such that the potential for caustic SCC is minimized and that applicants considered appropriate mitigation strategies, performance monitoring, and surveillance programs to minimize the potential for caustic SCC, to ensure that component integrity satisfies the design criteria.

Exothermic Reactivity with Water

The staff should evaluate whether the application adequately addressed the potential for molten sodium to react with water or moisture in the air to confirm that the design demonstrates that the potential for this phenomenon is minimized. Molten sodium undergoes a violent exothermic reaction on contact with water, which is a particular concern in the vicinity of steam generators (NRC, 2021e). Many such incidents from previously operating SFRs are documented (NRC, 2021e). The staff should verify that applicants minimize the potential for a sodium-water reaction through design, and that applicants considered appropriate mitigation strategies, performance monitoring, and surveillance programs to minimize the potential for contact between molten sodium and water or moisture in the air.

Impurity Effects on Corrosion

The staff should evaluate whether the application adequately addressed the temperature, flow rate, and impurity limits in the sodium coolant (notably, oxygen and carbon) since these parameters have a significant impact on the corrosion rate of metallic components in contact with the sodium coolant (Thorley and Tyzack, 1967; ANL, 2017; NRC, 2021e), which may lead to deleterious effects on structural performance due to increasing the likelihood of crack initiation or a reduction in strength or ductility. Studies conducted with varied levels of oxygen suggest that, to reduce oxidation and dissolution and maximize the lifetime of structural materials (mainly stainless steels) in SFRs, the oxygen level in sodium should be monitored and controlled to levels acceptable for a specific reactor design (Argonne, 1978, Hanford, 1980, NRC 2021e).

The staff should be aware that data from short-term (2,000 hours) static testing indicate that SiC/SiC may be resistant to corrosion from high-purity sodium (1 weight parts per million) at 550 degrees C, but the corrosion resistance decreases with an increasing concentration of oxygen in the sodium (Braun et al., 2021).

The staff should evaluate the applicant’s proposed mitigation strategies, performance monitoring, and surveillance programs to ensure that appropriate limits are set and maintained for key parameters for corrosion (e.g., flow rate and impurities, in particular oxygen).

Liquid Metal Embrittlement

The staff should evaluate whether applicants have adequately addressed liquid metal embrittlement (LME), as applicable, for metallic components in SFR. Some alloys are susceptible to LME in sodium, such as T91 steel (9Cr-1Mo-V) (Hemery et al., 2013). The staff should evaluate whether proposed mitigation, monitoring, and surveillance programs to manage LME are adequate.

Carburization and decarburization

The staff should be aware of the potential sources of sodium impurity corrosion mechanisms such as carbon in the sodium. Decarburization and carburization are both well documented in sodium reactors. Carbon impurity in the liquid sodium or transferred between materials in the liquid can induce carburization of heat exchangers, structural steels and control fuel rods in the reactor (NRC 2021e). The staff should evaluate that the potential for carburization and decarburization of structural alloys is controlled and to ensure that there are appropriate measures to maintain appropriate liquid sodium during plant design life.

Lead Coolant

A “lead-cooled” reactor may use lead (T_{melt}, 327.5 degrees C) or LBE alloy (T_{melt}, 123.5 degrees C) as the coolant. Metallic elements used in structural alloys, including iron, nickel, and chromium, are all soluble in lead or LBE, and that solubility in either coolant is a strong function of temperature (Ballinger and Lim, 2003; EPRI, 2019b). Above a certain temperature threshold, the use of typical ferritic and austenitic steels may require special treatments, such as alloying additions or coatings (EPRI, 2019b). It should be stressed that specific data on the environmental impact of molten lead and LBE on materials are not interchangeable since, for the same temperature, LBE is typically more corrosive than pure lead (NEA-OECD, 2015)

Below are additional degradation considerations likely to apply to lead-cooled liquid metal reactors. The staff should evaluate whether applicants have adequately addressed the following materials issues, including plans to monitor, evaluate, and mitigate degradation.

Corrosion at Higher Temperatures

Many alloys, including those approved for Section III-5 use, can experience high corrosion rates in a lead or LBE environment at higher temperatures, defined here as greater than 550 degrees C. For example, rapid corrosion of Type 316 stainless steel occurs above 550 degrees C even with tight oxygen control because of the transition from a protective to a nonprotective oxide (EPRI, 2019b). The staff should evaluate an applicant’s supporting test data over the entire range of operating temperatures to ensure that the designer has adequately characterized how the coolant may affect the mechanical properties of materials, including material susceptibility to LME (Gorse et al., 2011).

Effect of Flow Velocity

Increasing coolant flow velocities increase the effects of corrosion and should be considered (IRSN, 2012; Ballinger, and Lim, 2003). Studies have shown that decreasing flow velocity can control erosion for lead and LBE (Allen T.R. and Crawford D.C., 2007) and for pure lead coolant (Vogt, J.B. and Proriol Serre, I., 2021). Limiting flow velocities are not absolute but are temperature- and material-dependent. Higher flow velocities may be acceptable, especially for lower operating temperatures. The staff should review temperature, flow velocities and dissolved oxygen to accurately consider the combined effects of erosion and corrosion.

Liquid Metal Embrittlement

The staff should confirm that the applicant has adequately addressed the potential for LME of alloys used in lead-cooled reactors. LME is characterized by significant loss of ductility, caused by embrittlement of the grain boundaries of the solid alloy component and can also reduce creep life in some alloys. LME can be severe, depending on the alloy, operating temperature, and stress level of the affected components (EPRI, 2019b; OECD, 2007; Gorse et al., 2011).

Exposure to a lead or LBE environment has been shown to degrade the mechanical properties of some alloys, including ductility, fatigue resistance, and creep life. Ferritic/martensitic steels such as T91 (9Cr-1Mo-V) are more severely affected than austenitic steels (Type 316L) (Gorse et al., 2011). The staff should evaluate an applicant's material selection and supporting data to ensure that the potential effects of the lead or LBE environment on mechanical properties have been adequately addressed.

The staff should also confirm that the applicant has adequately addressed the effects of previous plastic deformation (e.g., cold work), which may affect the corrosion resistance of an alloy. The severity of dissolution corrosion attack in Type 316L stainless steel was found, in LBE coolant, to increase with increasing percentages of cold work (Klok et al., 2017).

The staff should verify that the applicant considered appropriate mitigation strategies, performance monitoring, and surveillance programs to minimize the potential for LME such that component integrity satisfies the design criteria.

Nonmetallic Materials

SiC/SiC has shown resistance to liquid metal corrosion up to 550 degrees C in 2000 hr corrosion tests in flowing liquid LBE (Takahashi, M. and Kondo, M.). Since experience with nonmetallics in lead or LBE environments is very limited, the staff should confirm that any use of nonmetallic materials in lead or LBE environments is supported by test data for the materials of interest in the relevant environment.

Oxygen Control

The corrosion potential of alloys in lead- and LBE-cooled fast reactors is highly dependent on temperature and the dissolved oxygen concentration in the coolants (EPRI, 2019b; Klok et al, 2018. Oxygen control is an important technique to ensure satisfactory performance of structural materials in lead- and LBE-cooled reactors. This technique, widely used in lead-based test facilities worldwide (Tarantino M., et al., 2021), consists of maintaining the oxygen concentration in the coolant within controlled limits. Corrosion rates at temperatures below 450 degrees Celsius are very low, and satisfactory operation in this temperature range can be achieved using many materials, including stainless steels and alloy steels (Ballinger and Lim, 2003). Strict oxygen control is necessary over the relevant range of temperatures and over the entire geometry of the coolant system, including local pockets or regions of off-chemistry coolant anywhere in the system, to maintain the protective oxide layer and avoid dissolution of alloying elements (EPRI, 2019b; Ballinger and Lim, 2003).

If oxygen concentration exceeds the solubility limits in the lead or LBE coolant, precipitation of lead oxide can occur, which can cause clogging of heat exchangers, as well as other detrimental effects on systems (OECD, 2007; IRSN, 2012). The staff should evaluate whether applicants considered appropriate mitigation strategies, performance monitoring, and surveillance programs to ensure that the dissolved oxygen content in the lead or LBE coolant is controlled so that component integrity satisfies the design criteria.

Reactor-Specific Guidance, Part 3: High-Temperature Gas Reactor

High-temperature gas-cooled reactors can use helium or carbon dioxide (CO_2) coolant; however, reactors that use CO_2 as the coolant, such as the Advanced Gas Reactor in the United Kingdom, are not currently expected to be deployed in the United States. Therefore, the following only addresses additional degradation considerations that are likely to apply to helium-cooled high-temperature gas-cooled reactors. Helium-cooled reactor designs under consideration in the United States include the high-temperature gas-cooled reactor (HTGR), the very high-temperature gas-cooled reactor, and the gas-cooled fast reactor (GFR), described in EPRI, 2020a. Common features of these designs include a reactor outlet temperature greater than or equal to 700 degrees Celsius. There is considerable operating experience from previous gas-cooled reactors operating in the United States and overseas (summarized in INL, 2011, and NRC, 2019). All these reactor designs are helium cooled, with the exception of the Advanced Gas Reactor and Magnox reactors in the United Kingdom. This section offers details on the design and/or environment-specific aspects of the general degradation mechanisms described in the "General Degradation Mechanisms" section above. The staff should evaluate whether applicants have adequately addressed the materials issues discussed below.

Creep-Rupture Strength

Service in a helium coolant environment has been shown to reduce the creep-rupture strength of structural alloys, in some cases resulting in lower creep-rupture strength than specified in ASME Code Section III-5 (Kim et al., 2013; Corwin, et al., 2008; NRC, 2021c). The staff should ensure that the potential for reduced creep-rupture strength within a helium coolant is accounted for in design analyses. The staff should confirm that applicants also considered appropriate mitigation strategies, performance monitoring, and surveillance programs to ensure that SSCs that could be impacted by lowered creep-rupture strength continue to satisfy the design criteria.

Emissivity

Emissivity is important in calculating heat transfer during operation and accident scenarios, and generally, higher emissivity is desired to assist in radiating heat (NRC, 2021c). Surface roughness can affect emissivity. In addition, the thermally grown surface oxide or carbide can affect emissivity on both the inside and outside of HTGR RPV. Within the RPV and primary loop SSCs, chemistry of the helium environment can have a significant effect on emissivity that should be accounted for in heat transfer calculations (NRC, 2021c). The staff should be aware of the potential for impurities in the helium coolant to affect the emissivity of structural alloys, as well as oxidizing impurities and abrasion or coating of metallic surfaces by graphite dust, which are other possible mechanisms for emissivity changes (NRC, 2021c). The staff should confirm that applicants also considered appropriate mitigation strategies, performance monitoring, and surveillance programs to ensure that SSCs continue to satisfy the design criteria.

Graphite

The staff should confirm that test data used to measure the coefficient of friction for graphite were gathered under conditions representative of operating temperatures and impurities in the coolant. The staff should be aware that the coefficient of friction for graphite is dependent on the graphite grade, temperature, and coolant impurities. The staff should be aware that impurities in the coolant have the potential to decrease the coefficient of friction (NRC, 2021g).

Graphite Dust

The staff should verify that applicants have adequately addressed the impact of graphite dust and debris in the coolant loop, which can be produced from the contact and movement of the pebbles or movement of the graphite blocks caused by temperature gradients, coolant flow, or vibrations (NRC, 2019). Graphite dust accumulations can decrease the efficiency of heat exchanger piping, hinder complete movement of the fuel or the control rod, and agglomerate on piping, clogging the flow of helium (NRC, 2002). Operational experience has demonstrated that graphite dust can also abrade piston rings in helium gas circulators, creating more dust in the primary loop and degrading the performance of the helium gas compressors (NRC, 2019). The staff should also be aware that graphite dust can carry absorbed fission products if fuel failure has occurred. The staff should confirm that applicants also considered appropriate mitigation strategies, performance monitoring, and surveillance programs to ensure that graphite dust is kept at acceptable levels so that SSCs continue to satisfy the design criteria.

Helium Impurities

Many operational issues in HTGRs have resulted from moisture intrusion into the helium coolant (NRC, 2019). The staff should therefore carefully evaluate the design aspects or operating practices that control moisture ingress. The main impurities present in helium coolant are water (H_2O), carbon monoxide (CO), methane (CH_4), hydrogen (H_2), and nitrogen (N_2). H_2O and CO affect oxidation, CO and CH_4 affect carburization, and H_2O affects decarburization (NRC, 2003; Sridharan, 2019). An Idaho National Laboratory (INL) report shows typical concentrations of these impurities in previously operating VHTRs (INL, 2006).

The staff should be aware of the potential sources and mechanisms of formation of impurities in the helium coolant. H_2O and O_2 present in the helium react with hot graphite in the core to form CO and H_2. CO_2 degassing from graphite also converts to CO. Corrosion reactions with alloys may also produce H_2 and CO. CH_4 can come from the leakage of oils (such as lubricants for circulators) or from the radiolytic reaction of H_2 with graphite (NRC, 2003; Sridharan, 2019). The staff should identify potential sources of impurities that could be introduced to the gas based on the specific design.

The staff should evaluate whether there is a favorable environment that leads to a stable oxide film and stable internal carbides (INL, 2006) and avoids excessive carburization, surface carburization, and decarburization. Other environmental factors to evaluate are the effects of temperature, alloy composition, and other impurities (NRC, 2021c). Figure 7 in NRC 2021c shows a schematic of favorable coolant gas characteristics to avoid rapid carburization or decarburization. Carburization can increase creep strength but decrease ductility, while decarburization can decrease lifetime by removing carbide strengthening phases. The staff should evaluate the coolant gas composition to ensure that the potential for carburization, decarburization, and oxidation of structural alloys is controlled and to ensure that there are appropriate measures to maintain appropriate gas composition during plant design life. HTGRs operated to date have maintained the total impurity levels in the helium below 10 parts per million to minimize these effects (INL, 2006).

Silicon Carbide and Silicon Carbide Composites

The staff should evaluate the use of SiC and SiC/SiC composites in HTGRs and consider the potential sources and effects of coolant impurities on these materials. For example, SiC and SiC/SiC may be susceptible to long term degradation under low partial pressures of oxygen (Choi, H., et al., 2021; EPRI, 2020a).

Lubricants

The staff should evaluate the use of oil lubricants in HTRs. Operational experience with HTRs has repeatedly demonstrated that coolant loops in different HTRs have been contaminated with oil-based lubricants (NRC, 2019), with deleterious impacts on the coolant purity.

IMPLEMENTATION

The staff will use the information discussed in this ISG to review non-LWR applications for construction permits and operating licenses under 10 CFR Part 50 and combined licenses, standard design approvals, design certifications and manufacturing licenses under 10 CFR Part 52 that propose to use materials allowed under ASME Section III, Division 5.

BACKFITTING, ISSUE FINALITY, AND FORWARD FITTING DISCUSSION

The NRC staff may use DANU-ISG-2023-01 as a reference in its regulatory processes, such as licensing, inspection, or enforcement. However, the NRC staff does not intend to use the guidance in this ISG to support NRC staff actions in a manner that would constitute backfitting as that term is defined in 10 CFR 50.109, "Backfitting," and as described in NRC Management Directive 8.4, "Management of Backfitting, Forward Fitting, Issue Finality, and Information Requests." The staff also does not intend to use the guidance to support NRC staff actions in a manner that constitutes forward fitting as that term is defined and described in Management Directive 8.4. If a licensee believes that the NRC is using this ISG in a manner inconsistent with the discussion in this paragraph, then the licensee may file a backfitting or forward fitting appeal with the NRC in accordance with the process in Management Directive 8.4.

CONGRESSIONAL REVIEW ACT

This ISG is a rule as defined in the Congressional Review Act (5 U.S.C. §§ 801–808). However, the Office of Management and Budget has not found it to be a major rule as defined in the Congressional Review Act.

PAPERWORK REDUCTION ACT

This ISG provides voluntary guidance for implementing the mandatory information collections in 10 CFR Parts 50 and 52 that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et. seq.). These information collections were approved by the Office of Management and Budget (OMB), approval numbers 3150-0011 and 3150-0151. Send comments regarding this information collection to the FOIA, Library, and Information Collections Branch (T6-A10M), U.S. Nuclear Regulatory Commission, Washington, DC 20555 0001, or by e-mail to Infocollects.Resource@nrc.gov, and to the OMB reviewer at: OMB Office of Information and Regulatory Affairs (3150-0011 and 3150-0151), Attn: Desk

Officer for the Nuclear Regulatory Commission, 725 17th Street, NW Washington, DC 20503.

FINAL RESOLUTION

The staff will transition the information and guidance in this ISG into RG 1.87 or NUREG series, as appropriate. Following the transition of all pertinent information and guidance in this document into the RG or NUREG series, or other appropriate guidance, this ISG will be closed.

APPENDICES

A Analysis of Public Comments on Interim Staff Guidance (ISG): Material Compatibility for Non-Light Water Reactors
B References

DANU-ISG-2023-01 APPENDIX A

Analysis of Public Comments on Interim Staff Guidance (ISG): Material Compatibility for Non-Light Water Reactors

Comments on the subject draft Interim Staff Guidance (ISG) are available electronically at the U.S. Nuclear Regulatory Commission's (NRC's) electronic Reading Room at http://www.nrc.gov/reading-rm/adams.html. From this page, the public can access the Agencywide Documents Access and Management System (ADAMS), which provides text and image files of the NRC's public documents. The following table lists the comments the NRC received on the draft ISG.

Letter Number	ADAMS Accession No.	Commenter Affiliation	Commenter Name
1	ML23069A091	Public	A. Thomas Roberts
2	ML23076A284	Engie-Tractebel	Michel Desmet
3	ML23130A195	Hybrid Power Technologies LLC	Michael Keller
4	ML23130A196 ML23130A202 ML23130A205 ML23130A206 ML23130A207	Public	Anonymous
5	ML23130A208	NEI	Mark Richter
6	ML23130A209 ML23130A210 ML23130A211 ML23130A212	Public	Anonymous
7	ML23130A213	EPRI	Chris Wax
8	ML23194A112	INL	Sam Sham

The original comment as written by the commenter in its letter above is listed first, followed by the NRC staff's response.

Letter 1— Allen Thomas Roberts

<u>Comment No. 1-1</u>

While some advanced reactor designs operate at high temperatures and are expected to be subject to the listed degradation mechanisms (DMs), some also operate at very low pressures.

Hence pressure loads might sometimes be negligible compared to potential seismic loadings.

While seismic loading is accounted for under ASME III Division 5, the seismic load effects on some materials and the ISG identified degradation mechanisms may be lacking sufficient guidance for license applicants to be mindful of what the USNRC may expect especially in term of functionality.

For example, a prismatic core of graphite may experience a critical turnaround dose level and the graphite strength will increase gradually up to that turnaround dose but then rapidly decrease in strength after turnaround. However, the diminished in-situ strength of a prismatic graphite core following turn around would not likely prevent it from serving its control rod bank shutdown function, provided there is no misalignment of prismatic blocks that would preclude control rod insertion that resulted from a seismic load.

While there are several listed DMs that may or may not have functionality impacts because of seismic loading versus traditionally emphasized pressure loads, it is recommended that a caution be provided in the ISG regarding DMs that may be functionally affected in concert with seismic loadings.

NRC Response

The staff made no changes to the ISG based on this comment. Component design, including seismic load effects, is outside of the scope of this ISG.

Comment No. 1-2

Advanced Manufacturing Technology (AMT) produced products are anticipated to be utilized in several advanced reactors. While these materials may exhibit some mechanical and chemical properties comparable to their traditionally produced counterpart products (e.g., forged, cast, etc.), the microstructures for these materials are likely to be very different.

Consequently, inspections (e.g., NDE) and monitoring of SSCs produced using AMT processes may result in the need to adjust and confirm the suitability of presently used NDE techniques (e.g., UT, ET, AE, etc.) assuming that traditional NDE methods are employed to monitor these AMT produced SSCs.

As cited in several studies, AMT examined material when UT examined results in different sound attenuation properties than their counterpart traditionally produced products. These potential differences in AMT material characteristics are the principal reason that ASME Section XI Division 2, Reliability Integrity Management (RIM), noted in this ISG, requires performance demonstration of all selected monitoring or NDE methodology for assessing potential deterioration of an SSC over its service life.

This performance demonstration (PD) and the derived numerical output is an essential element required by RIM, because the PD is factored back into the Reliability Target value(s) that is established and assigned to risk significant SSCs.

It is recommended that this ISG provide brief guidance on monitoring and NDE methodologies that might be anticipated to be utilized on AMT produced items. This guidance would not only serve the USNRC staff reviews but inform license applicants of matters they should consider.

NRC Response

The staff made no changes to the ISG based on this comment. The ISG provides guidance on evaluation of environmental degradation and materials issues. Guidance on specific monitoring and NDE methodologies is outside of scope for this ISG. The ISG is not intended to be prescriptive in methods to demonstrate acceptability of SSCs, traditionally fabricated or those fabricated from AMTs.

Comment No. 1-3

As an extension of Comment 2, the use of Silicon Carbide (SiC), Carbon-Carbon (C/C), and Silicon Carbide (SiC/SiC) composites in risk significant SSC is an area that should be cautioned for the need to conduct performance demonstration of monitoring and NDE methodologies proposed to be employed on over their service life.

NRC Response

The staff agrees with this comment but determined that changes to the document are out of scope of the ISG. Therefore, staff made no changes to the ISG based on this comment. Staff believes that demonstrating effectiveness of monitoring and NDE methodologies is important for all components, not just those fabricated with non-metallic materials.

Comment No. 1-4

As noted in the technology specific portions of the ISG, several reactor technologies are cited as being potentially susceptible to corrosion related mechanisms (e.g., Caustic Stress-Corrosion Cracking, Impurity Effects on Corrosion, etc.).

The ability to effectively provide monitoring or even NDE techniques for these DMs that might occur in risk significant SSCs should be highlighted as possibly needing consideration of a systematic approaches to monitor for these possible DMs (e.g., in-situ chemistry monitoring systems, installed transducers for wall thickness monitoring, etc.) rather than the traditionally focused NDE approaches to mostly look only at weld locations.

Providing this guidance would not only serve during the USNRC staff reviews but also inform licensee applicants of matters they should consider.

NRC Response

The staff agrees with this comment, in part. The ISG does not define specific monitoring or NDE technologies that should be implemented; however, providing examples would help clarify the intent of the phrase "performance monitoring." The following statement will be added to page 5, Q*ualification and Performance Monitoring,* "Examples of this type of performance monitoring that could be appropriate include chemistry, temperature, or flow monitoring, as well as wall thickness measurements."

Letter 2 – Engie-Tractebel

Comment No. 2-1

Page 9, "Thermal Fatigue and Transients", 1st paragraph, before last line: stripping to be corrected to striping.

NRC Response

The staff agrees with and has incorporated this comment.

<u>Comment No. 2-2</u>

Page 9, "General Materials Issues", 1st paragraph, 3rd line: 'the need to...' to be replaced by 'there is a need to...'?

<u>NRC Response</u>

The staff made no changes to the ISG based on this comment. The paragraph was edited in response to comment 7-5 and no longer contains this wording.

Letter 3 – Hybrid Power Technologies LLC

<u>Comment No. 3-1</u>

The proposed staff guidance clearly demonstrates that the NRC staff considers that the ASME Code does not adequately protect the public from hazardous radiation. The thin justification provided for the NRC staff's position is deeply troubling, particularly in light of the ASME Code's long history of successfully governing boilers, pressure vessels, and reactors. Stated somewhat differently, the NRC staff apparently does not consider that the ASME Code properly and proportionately provides for the Safety Related and allied tiered Important-to-Safety functions that form the backbone of the entire regulatory process. Rather than unilaterally imposing NRC Staff desires on licensees and applicants, the NRC Staff should collaborate with the ASME to reach a mutually acceptable code. The proposed staff guidance contains a number of technical considerations whose linkages to proportional risk-informed technical design considerations are not apparent. The staff's technical design considerations unilaterally impose wide ranging and obtuse technical justification requirements on reactor materials, irrespective of the proportional level-of-risk to fundamental safety functions, and ultimately the hazardous radiation risk to the public. The interests of the public and industry are better served by reliance on the well proven ASME Code because the code specifically considers, in-depth, items such as service conditions, proper design margins, proper design methods, and in-service inspections.

The ISG does not cite the specific section(s) of the code that form the basis for the NRC staff claims. In general, limitations on code rules are, in part, of the form

> *do not cover deterioration that may occur in service as a result of radiation effects, corrosion, erosion, thermal embrittlement or instability of the material. These effects shall be taken into account with a view to realizing the design or the specified life of the components and supports. The changes in properties of materials subjected to neutron radiation may be checked periodically by means of material surveillance programs.*

NRC Staff claim that the code does not address environmental conditions is simply not true.

The ASME Code has been developed through the collective historical wisdom of hundreds industry experts and firms integrally involved in the design, manufacture, construction, and operation of boilers, reactors, and pressure vessels. By contrast, the regulatory bureaucracy's expertise does not lie in these areas. That same observation applies to national laboratories. The NRC staff should comply with Congress' explicit direction contained in the Act by reaching agreement on a collaboratively developed ASME Code.

<u>NRC Response</u>

The staff disagrees and made no changes to the ISG based on this comment. It is expected that an applicant would use Division 5 for the design and qualification, as applicable, of materials, as conditioned by RG 1.87. This ISG is intended to provide staff guidance on addressing environmental effects that are not included in the scope of Division 5.

ASME Section III, Division 5, HAA-1130 “LIMITS OF THESE RULES” specifically states that Division 5 rules “…do not cover deterioration that may occur in service as a result of radiation effects, corrosion, erosion, thermal embrittlement, or instability of the material. These effects shall be taken into account with a view to realizing the design or the specified life of the components and supports. The changes in properties of materials subjected to neutron radiation may be checked periodically by means of material surveillance programs.”

<u>Comment No. 3-2</u>

Attempting to use lower-tier discretionary regulatory guidance to bludgeon highly technical justification requirements onto licensees and applicants is of highly questionable flexibility, practicality, and authority. The issue is particularly troubling for technical considerations well removed from materially impacting the public as a result of remotely likely hazardous radiation releases.

We have observed an NRC Staff propensity to use regulatory guides and similar lower tier documents to create new de facto technical requirements when the staff does not get their way during code and standard development activities. The areas of disagreements are generally associated with arcane technical considerations well removed from materially affecting fundamental nuclear safety concerns. With the passage of the Act, such behavior is not permissible.

In conjunction with the 10CFR53 development effort, we have previously formally expressed concerns over the NRC staff overriding codes and standards using lower tier regulatory guidance documents – see regulations.gov, comment section of docket NRC-2019-0062.

We note that the NRC staff was, in fact, part of the development process for the subject ASME Code and had ample opportunity to reach a collaboratively agreed to code. The ASME Code development process is fully transparent and unquestionably fair as all comments and concerns are forthrightly openly discussed and resolved. The ASME process stands in stark contrast with the NRC staff’s secretive regulatory approach, as amply demonstrated by the 10CFR53 process involving hundreds of pages of unresolved comments and concerns. In closing, in our opinion, the proposed guidance fails to properly comply with the Act in a number of key areas and is therefore of doubtful legality. The NRC staff should avoid using lower tier guidance to impose de facto technical justification requirements in an attempt to preempt the lawful domain of industry codes and standards. The document should be withdrawn and not issued.

<u>NRC Response</u>

The staff disagrees and made no changes to the ISG based on this comment. The staff have processes to endorse industry Codes, and recently completed its endorsement of the 2017 edition of ASME Section III, Division 5. In areas where there is not industry guidance, or areas where the NRC does not endorse industry guidance, NRC staff develops guidance documents such as ISGs and RGs to fill that void. In this instance, Division 5 does not provide guidance on environmental testing/compatibility, except for graphite, beyond stating that those effects should be taken into account. In addition, this is staff guidance and does not impose additional requirements on applicants beyond those in Division 5, as applicable.

Letter 4 – Anonymous

Comment No. 4-1

On May 22, 2007, OMB issued Memorandum M-07-16, Safeguarding Against and Responding to the Breach of Personally Identifiable Information, which required Federal agencies to publish a routine use for their systems of records specifically applying to the disclosure of information in connection with response and remedial efforts in the event of a breach of personally identifiable information. FWS published a notice in the Federal Register in 2008 to modify all FWS system of records by adding a routine use in their “ROUTINE USES” section to address the limited disclosure of records.

NRC Response

The staff made no changes to the ISG based on this comment as it is outside the scope of the ISG.

Letter 5 – NEI

Comment No. 5-1

Section: General

Comment/Basis: For performance monitoring and surveillance, would it be acceptable to have materials from the same heat tested in a simulated environment?

Recommendation: Allow materials from the same heat to be tested in a simulated environment to satisfy the surveillance requirement.

NRC Response

The staff made no changes to the ISG based on this comment. It is outside the scope of the ISG to make a general determination on acceptability of specific performance monitoring, surveillance, material qualification, or inspection methods.

Comment No. 5-2

Section: General

Comment/Basis: Several of the sections on degradation mechanisms end with the statement: “The staff should confirm that applicants also consider appropriate mitigation strategies, performance monitoring, and surveillance programs to ensure that SSCs affected by corrosion continue to satisfy the design criteria.”

Recommendation: It is suggested to be more specific on which “design criteria” the ISG refers to at the end, for example by saying “…to satisfy the principal design criteria.”

NRC Response

The staff made no changes to the ISG based on this comment. It is out of scope of this ISG to determine if appropriate principal design criteria have been determined by an applicant.

Comment No. 5-3

Section: Qualification and Performance Monitoring

Comment/Basis: The following paragraph mentions testing, but it does not specifically allow for the use of data from previous facilities within the same parametric operating envelope.

Materials qualification and monitoring programs should include testing conducted in an environment simulating the anticipated operating environment for the reactor, including chemical environment, temperatures, and irradiation. Testing should account for uncertainties in the environment, material composition, fabrication methods, and operating conditions. The scope of this testing should include safety-related component materials, safety-significant component materials, and as needed, non-safety related component materials whose failure could impact critical design functions. Testing should be conducted to determine if materials properties and allowable stresses meet applicable codes and standards or other design requirements. If necessary, appropriate reduction factors should be applied to the materials properties and allowable stresses from the applicable design codes and/or design specifications.

Recommendation: Guidance should be updated to allow for the use of data from previous facilities within the same parametric operating envelope.

NRC Response

The staff agrees, in part, with this comment. Historical data could be appropriate to use if directly applicable to supplement materials qualification and testing programs. The staff has updated the ISG to include the italicized text:

> “The selection of structural materials for the reactor design should consider effects on the materials properties and allowable stresses due to interactions with the operating environment. Materials qualification and monitoring programs should include testing conducted, *or use of historical data collected*, in an environment simulating the anticipated operating environment for the reactor, including chemical environment, temperatures, and irradiation. *It is incumbent upon the applicant to demonstrate that data is directly applicable to the plant design and environment.* Testing *or historical data* should account for uncertainties in the environment, material composition, fabrication methods, and operating conditions. The scope of this testing should include safety-related component materials, safety-significant component materials, and as needed, non-safety related component materials whose failure could impact critical design functions. Testing should be conducted to determine if materials properties and allowable stresses meet applicable codes and standards or other design requirements. If necessary, appropriate reduction factors should be applied to the materials properties and allowable stresses from the applicable design codes and/or design specifications.”

Comment No. 5-4

Section: Qualification and Performance Monitoring, page 6, first paragraph

Comment/Basis: The ISG says: “In the meantime, staff should evaluate whether applicants have adequately addressed the following general degradation mechanisms for various reactor environments.”

Recommendation: It is suggested that the extent to which such degradation mechanisms should be addressed should be commensurate with their safety significance. A possible rewording could be: “In the meantime, staff should evaluate whether applicants have adequately addressed the following general degradation mechanisms for various reactor environments, to an extent which should be commensurate with the safety significance of the degradation mechanism.”

NRC Response

The staff has made no changes to the ISG based on this comment. The ISG is sufficiently clear that the scope of the ISG is for safety-related component materials, safety-significant component materials, and as needed, non-safety related component materials whose failure could impact critical design functions.

Comment No. 5-5

Section: Qualification and Performance Monitoring, page 6, last paragraph

Comment/Basis: When saying: “Erosion products from SSCs have the potential for depositing elsewhere in the coolant flow path, affecting coolant flow patterns and local heat transfer properties,” it is suggested to also say that these erosion products may undergo activation, thus contributing to the activity of the coolant itself.

Recommendation: A possible wording could be: “In addition to potentially undergoing activation thus contributing to the coolant’s activation level, erosion products from SSCs have the potential…. heat transfer properties.”

NRC Response

The staff agrees with this clarification and has incorporated the comment.

Comment No. 5-6

Section: Qualification and Performance Monitoring, page 9, first paragraph

Comment/Basis: Correct typo: “striping” in place of “stripping” within the sentence: “The staff should ensure that very high cycle fatigue due to thermal stripping has been adequately addressed by the applicant.”

Recommendation: none

NRC Response

The staff agrees with and has incorporated the comment.

Comment No. 5-7

Section: Qualification and Performance Monitoring, page 9, end of “Thermal and Fatigue Transients”

Comment/Basis: It is suggested to add “whenever applicable” within the sentence: “The staff should verify that synergistic effects of thermal fatigue, vibratory fatigue, and creep-fatigue are addressed by the applicant” so that it reads: “The staff should verify that, whenever applicable, synergistic effects of thermal fatigue, vibratory fatigue, and creep-fatigue are addressed by the applicant”

Recommendation: none

NRC Response

The staff agrees with and has incorporated the comment.

Comment No. 5-8

Section: Qualification and Performance Monitoring, page 9, “Wear/Fretting”

Comment/Basis: It is suggested to clarify the following paragraph: “The staff should consider the potential impacts of the specific coolant environment on wear and fretting, particularly in heat exchangers in steam generators. Depending on the reactor design, the interaction between the coolants in the primary, secondary, and steam generating loops may have adverse consequences for the reactor with regard to wear and fretting.”

Recommendation: Specifically, “in heat exchangers” and “in steam generators” seem to be a duplication, and the latter can be deleted. Moreover, when speaking about “interaction between coolants,” it is not clear whether the subject interaction is between the coolants, or between the coolants and the heat exchanger structures. This part would benefit from a rewording.

NRC Response

The staff agrees with this comment and has clarified the intent of the paragraph, which now states, “heat exchangers and steam generators” and also clarifies the interaction between the coolants is as a result of wear and fretting.

Comment No. 5-9

Section: Page 9, first paragraph under “General Materials Issues”

Comment/Basis: The ISG says: “The staff should evaluate whether applicants have adequately addressed the following design neutral materials issues as appropriate for the application and design.” It is suggested to indicate that the extent with which such design neutral material issues should be addressed should be commensurate with their safety significance.

Recommendation: A possible rewording could be: “The staff should evaluate whether applicants have adequately addressed the following design-neutral materials issues as appropriate for the application and design, to an extent which should be commensurate with the safety significance of each issue.”

NRC Response

The staff has made no changes to the ISG based on this comment. The ISG is sufficiently clear in the Discussion that the scope of the ISG is for safety-related component materials, safety-significant component materials, and as needed, non-safety related component materials whose failure could impact critical design functions.

Comment No. 5-10

Section: Page 11

Comment/Basis: Graphite-salt compatibility considerations include fluorination of the graphite and formation of carbides (uranium carbide, chromium carbide, and others), as well as potential infiltration of molten salt into the graphite (ORNL, 2021a.)

Recommendation: Comment 1: ORNL, 2021a is missing in the draft ISG as a reference.

Comment 2: The fact that fluorination is the first stated compatibility issue might be concerning. There is no relevant data that fluorination is a real thing for all engineering purposes (although it is mentioned in the literature). At most, this should be demonstrated with a test program and not require monitoring.

Comment 3: Formation of carbides is not a concern for the graphite itself (see comment 1 below for Page 12).

Comment 4: Infiltration should be demonstrated with a test program and not require monitoring. Dispensing with monitoring might require more data from the test program to show infiltration is not occurring and/or effective infiltration is not safety significant. Infiltration is a greater concern for fuel salt MSRs because the fissile material in the salt could create localized high temperature regions, e.g., "hotspots" and xenon accumulation.

NRC Response

Comment 1: The staff has corrected a typographical error regarding the misattributed reference.

Comment 2-3: The staff disagrees with this comment. Fluorination and the formation of carbides are discussed in the corrected reference, which is why they are included as possible mechanisms in the ISG. Therefore, no changes have been made to the ISG to address these comments.

Comment 4: While the staff agree testing programs might be appropriate for infiltration, the staff finds that monitoring and NDE could also be demonstrated to be viable methods to ensure a component is capable of meeting design criteria. Staff have made the following edits to clarify the section: “The staff should confirm that graphite qualification, monitoring, surveillance, or inspection programs address any potential chemical compatibility issues, as applicable.”

Comment No. 5-11

Section: Page 11, first paragraph under "Reactor Specific Guidance, Part 1: Molten Salt Reactors"

Comment/Basis: It is recommended to correct the definition of MSRs operating with solid fuel, as the ISG indicates that in these MSRs "the molten salt coolant has relatively small amounts of fissile material and fission products," which is not true as the fissile material is contained within the fuel, not the coolant. In addition, when referring to TRISO later in the same paragraph, it is suggested to indicate that, although dominant, this is "just" an example of solid fuel used in solid-fuel MSRs.

Recommendation: none.

NRC Response

The staff agrees with and has incorporated the comment. The section now states: "In a solid-fuel MSR, the fissile material and fission products are typically contained within a TRISO fuel particle…additionally, relatively small quantities of fission products may be present in the molten salt coolant".

Comment No. 5-12

Section: General Degradation Mechanisms - Irradiation

Comment/Basis: Guidance notes that: "The staff should evaluate data on the effects of neutron irradiation on materials, including mechanisms such as irradiation assisted creep, irradiation embrittlement, irradiation-assisted SCC, and decreased resistance to oxidation."

Recommendation: Given that III-5 does not provide specific acceptable means to account for irradiation effects on structural material properties, the guidance should be updated to provide additional detail on staff expectations for review or acceptable means to account for irradiation effects on structural material properties.

NRC Response

The staff disagrees this is a necessary addition and made no changes to the ISG resulting from this comment. Given the wide variety of operating conditions, materials and coolants in high temperature reactors and the current state of knowledge, it is not practical to provide specific guidance to resolve this comment. The ISG does not provide specific guidance for any of the other degradation mechanisms or materials issues.

Comment No. 5-13

Section: General Degradation Mechanisms: Silicon Carbide

Comment/Basis:

1. SiC is captured in the General Materials Issues section with the main takeaway that all SiC types should be qualified separately.

2. For the reactor specific guidance sections, molten salt and liquid metal reactors (sodium and Lead coolant) both specifically call out SiC but strangely there is no SiC reference under HTGRs. Given this is where GA-EMS is most focused, perhaps we should look to add something in there.
    a. Under the section “Reactor Specific Guidance, Part 3: High Temperature Gas Reactor”, the document states that “NRC is not aware of any current plans to deploy GFR reactors in the Unites States, so this section does not address materials concerns for GFRs.”
3. Not all of the degradation mechanisms are broadly applicable to all candidate reactor materials. For instance, stress relaxation cracking is an identified mechanism in heat affected zone of alloy welds but is not an expected damage mechanism in silicon carbide material.

Recommendation:

1. It should be acknowledged that degradation mechanisms are fundamentally different in SiC-SiC composites compared to metals (Jacobsen, GA-EMS, JNM, 452, p125-132, 2014). The staff should be aware that different mechanisms (e.g. - matrix cracking, fiber sliding) and different analytical techniques (e.g. – Weibull analysis) must be considered to account for the stochastic behavior of SiC materials.
2. This is an issue that needs to be resolved. GA-EMS is developing the Fast Modular Reactor with the intent to deploy in the US, and this design leverages non-metallic materials, specifically SiC/SiC composite due to its demonstrated high temperature performance and compatibility with Helium coolant (Choi, GA-EMS, ANS Transactions, 124, p454-456, 2021). As is written above in the molten salt and metal coolant sections, the staff should be aware of the potential sources and impacts of impurities in the Helium and the effects these have on SiC/SiC performance and degradation mechanisms.
3. There should be an avenue to specify if certain mechanisms aren’t applicable to a material system or plant design, in addition to the stated feasibility of adding additional mechanisms.

NRC Response

Comment 1: The staff made no changes to the ISG based on this comment. The staff agrees the mechanical characteristics of nonmetallic matrix composites are uniquely different from metals, however, the staff does not see the need to explicitly state this in the ISG. The ISG contains a reference to the 2021 edition of ASME Section III, Division 5 which acknowledges the differences between nonmetallic matrix composites and metals.

Comment 2: The staff agrees that use of SiC and composites are appropriate to consider for unique performance and compatibility in HTGRs and has added a section on silicon carbide to this section of the ISG. The staff has incorporated comment 2a into the ISG.

Comment 3: The staff disagrees and made no changes to the ISG based on this comment. For all degradation mechanisms identified in the ISG, it is incumbent on the applicant to determine and justify if they are or are not applicable. Further, it is generally not possible to eliminate a priori particular degradation mechanisms without knowledge of the design, environment, and material combination.

Comment No. 5-14

Section: Page 12

Comment/Basis: The staff should evaluate whether the application adequately addressed the potential formation of uranium and other metal carbides on graphite, and subsequent deleterious effects on reactor materials (EPRI, 2019a.)

Recommendation: Per the EPRI report, the only concern with metal carbide forming on graphite seems to be related to corrosion of metals, not related to degradation of the graphite itself. Also, the concern with uranium carbide for fuel-salt MSRs is related to “nuclear performance” (neutronics), not graphite degradation. This evaluation should be removed as a compatibility issue.

NRC Response

The staff disagrees with the comment and added a reference to NRC 2021d supporting the staff’s position that formation of uranium and other metal carbides on graphite should be addressed in an application.

Comment No. 5-15

Section: Liquid Metal Reactors – Caustic Stress Corrosion Cracking

Comment/Basis: Although it is noted that most steam generators, both tubes and shell, are made of ferritic steels, austenitic stainless steels have been used successfully in previous sodium fast reactor steam generators (e.g., EBR-II and the Prototype Fast Reactor (PFR) operated on the Dounreay site).

Recommendation: Guidance should be updated to reflect this operational experience.

NRC Response

The staff has modified the language in response to the comment to highlight the importance of operating experience, particularly with the use of austenitic steels for steam generator tubing in the French Phoénix reactor. The staff found multiple references (listed below) stating the EBR-II steam generator tube material were ferritic steel alloyed with chromium and molybdenum and so, disagrees with the comment. The staff notes the failure of approximately 40 steam generator tubes in the Prototype Fast Reactor (PFR) caused by the sodium-water reaction. The staff notes, however, the source of the damage to the PFR steam generator tubes was caused by residual weld stresses.

Buschman H.W., Penney H.W., and Longua K.J., "Operating Experience of the EBR-II Intermediate Heat Exchanger in the Steam Generator System,” ASME/IEEE Joint Power Generation Conference in Indianapolis, Indiana, September 25-29, 1983.

Buschman H.W., Penney H.W., and Longua K.J., "The EBR-II Steam Generating System – Operation, Maintenance, and Inspection," IAEA-IWGFR Specialists' Meeting on Maintenance of LMFBR Steam Generators in Oarai Japan, June 4-8, 1984.

International Atomic Energy Commission, “Fast Reactor Database: 2006 Update,” IAEA-TECDOC-1531, 2006.

Comment No. 5-16

Section: Liquid Metal Reactors – Impurity effects on corrosion

Comment/Basis: Successfully operated sodium fast reactors (e.g., the Experimental Breeder Reactor-II) and standards developed for SFR systems have established maximum acceptable oxygen levels in sodium of 2 ppm.

“EBR-II Operating Experience,” Section 5.2, “Source Rate of Impurities,” (1978) notes that “EBR-II operating limits for primary sodium are 2.0 ppm oxygen and 200 ppb hydrogen. Normal concentrations are ~0.8 ppm oxygen and ~90 ppb hydrogen.” RDT A 1-5T, “Purity Requirements for Operating Sodium Reactor Systems,” (1973) specifies an oxygen concentration limit of up to 2.0 ppm for hot leg temperatures >800 F.

Recommendation: While it is noted that higher oxygen concentration has been seen to increase the corrosion rates of steels in a sodium environment, the guidance should be updated to reflect that oxygen levels of 2 ppm have been shown to be acceptable.

NRC Response

The staff disagrees that a specific oxygen level should be provided; however, found it appropriate to make the following change, as indicated in italics, to the ISG for added clarity:

> “Studies conducted with varied levels of oxygen suggest that, to reduce oxidation and dissolution and maximize the lifetime of structural materials (mainly stainless steels) in SFRs, the oxygen level in sodium should be monitored and controlled *to levels acceptable for a specific reactor design* (Argonne, 1978, Hanford 1980, NRC 2021e).”

Comment No. 5-17

Section: Page 13, first paragraph under Liquid Metal Reactors

Comment/Basis: When introducing LBE it is suggested to indicate the composition of this eutectic, for example by adding it at the end of the sentence: “Liquid metal reactors are characterized by their operation at or near ambient pressure using a fast neutron spectrum in which the fuel, with metallic cladding, is cooled by liquid sodium, lead, or the lead-bismuth eutectic (LBE, 44.5 wt% Pb and 55.5 wt% Bi).” This is to clarify that the composition of this eutectic is very far from pure Pb.
Recommendation: none

NRC Response

The staff agrees that the addition provides clarity and incorporated the comment.

Comment No. 5-18

Section: Page 13, first paragraph under Liquid Metal Reactors

Comment/Basis: When saying: “To date, operational experience with LFRs is limited to propulsion nuclear reactors in Alfa-class submarines operated by the Soviet Union from 1967–1983,” it is suggested to specify that these reactors were LBE-cooled. Recommendation: A possible rewording could be: “To date, operational experience with LFRs is limited to LBE-cooled propulsion nuclear reactors in Alfa-class submarines operated by the Soviet Union from 1967–1983.”

NRC Response

The staff agrees with and incorporated the comment with a minor modification and added an additional reference.

Comment No. 5-19

Section: Page 15, first paragraph under “lead coolant”

Comment/Basis: The ISG says: “As a result, use of typical ferritic and austenitic steels requires special treatments, such as alloying additions or coatings (EPRI, 2019b).”

Recommendation: It is recommended to correct this sentence, as the need for “special treatments” is not absolute but depends on the temperature. Specifically, typical steels do not require special treatments when the temperature is below approx. 480C, which is the operating temperature for internals operating at cold leg temperature. A possible rewording could be: “As a result, when the temperature is above approximately 480°C, use of typical ferritic and austenitic steels require special treatments, such as alloying additions or coatings (EPRI, 2019b).”

NRC Response

The staff agrees there is a temperature relationship and incorporated the comment, with modification. The ISG now states, “Above a certain temperature threshold, the use of typical ferritic and austenitic steels may require special treatments, such as alloying additions or coatings (EPRI, 2019b).”

Comment No. 5-20

Section: Page 15, end of first paragraph under “lead coolant”

Comment/Basis: It is suggested to more strongly emphasize the (correct) statement:” Specific data of the environmental impacts of molten lead and LBE on materials are not interchangeable,” as the two are often confused. A proposed rewording is: “It should be stressed that specific data on the environmental impact of molten lead and LBE on materials are not interchangeable since, for the same temperature, LBE is typically more corrosive than pure lead (Ref. X).” where Ref. X is: NEA-OECD, Handbook on Lead-bismuth Eutectic Alloy and Lead Properties, Materials Compatibility, Thermal-hydraulics and Technologies. 2015 edition

Recommendation: none

NRC Response

The staff agrees with and incorporated the comment.

Comment No. 5-21

Section: Page 15, second paragraph under "lead coolant"

Comment/Basis: The ISG says: "The staff should evaluate whether applicants have adequately addressed the following materials issues, including plans to monitor, evaluate, and mitigate degradation." It is suggested to add that the extent to which this is addressed should be commensurate with the safety significance of the degradation mechanism.

Recommendation: A proposed rewording could be: "The staff should evaluate whether applicants have adequately addressed the following materials issues, including plans to monitor, evaluate, and mitigate degradation, in a way commensurate with the safety significance associated with each degradation mechanism."

NRC Response

The staff has made no changes to the ISG based on this comment. The ISG is sufficiently clear in the Discussion that the scope of the ISG is for safety-related component materials, safety-significant component materials, and as needed, non-safety related component materials whose failure could impact critical design functions.

Comment No. 5-22

Section: Page 15, last paragraph

Comment/Basis: It is suggested to add an indication of the temperature range and additional references at the end of the following sentence:

"Non-code-qualified materials such as alumina forming or aluminum-coated stainless steels and silicon-enriched stainless steels may provide enhanced corrosion resistance in LBE and lead coolants at high temperatures (EPRI, 2019b; OECD, 2007; Ballinger and Lim, 2003)", so that it reads:

"Non-code-qualified materials (…) in LBE and lead coolants at high temperatures up to at least 700-750°C (EPRI, 2019b; OECD, 2007; Ballinger and Lim, 2003; Ref. A, Ref. B, Ref. C, Ref. D, Ref. E)" where the references are:

Ref. A: F. García Ferré, et al., "Corrosion and radiation resistant nanoceramic coatings for lead fast reactors," Corrosion Science, 124 (2017) 80-92.

Ref. B: DOMSTEDT, P., LUNDBERG, M., SZAKALOS, P., Corrosion Studies of Low-Alloyed FeCrAl Steels in Liquid Lead at 750 °C. Oxidation of Metals (2019) 91:511–524. https://doi.org/10.1007/s11085-019- 09896-z

Ref. C: DOMSTEDT, P., et. al., (2020), Corrosion studies of a low alloyed Fe–10Cr–4Al steel exposed in liquid Pb at very high temperatures. Journal of Nuclear Materials. 531. 152022. 10.1016/j.jnucmat.2020.152022.

Ref. D: CHEN, L., et al., Investigation of microstructure and liquid lead corrosion behavior of a Fe-18Ni-16Cr-4Al base alumina-forming austenitic stainless steel. Mater. Res. Express 7 (2020) 026533. https://doi.org/10.1088/2053-1591/ab71d1

Ref. E: PINT, B.A., SU, Y.F., BRADY, M.P., et. al., Compatibility of Alumina-Forming Austenitic Steels in Static and Flowing Pb. JOM 73, 4016–4022 (2021). https://doi.org/10.1007/s11837-021-04961-y

Recommendation: none

NRC Response

The staff has deleted this section as a result of Comment 7-9.

Comment No. 5-23

Section: Page 16, “Lead Erosion”

Comment/Basis: It is recommended to reword the statement: “Lead is highly eroding, and for this reason, the flow velocity should be limited (IRSN, 2012; Ballinger and Lim, 2003)” as neither of the references provided gives evidence that “lead is highly eroding.”. While it is true that the lead flow velocity must be limited to prevent erosion effects, as written the text is misleading.

Recommendation: It is suggested to reword as: “At high lead (or LBE) flow velocities the effect of erosion, in addition to corrosion, should be considered. Even though the flow velocity limit is not absolute but temperature- and material-dependent, common practice is to maintain the velocity of lead-based coolants in high temperature regions of the reactor coolant system, such as the core, below approximately 2 m/s both for LBE (Ref. A) and for pure lead coolant (Ref. B). Higher velocities may be acceptable especially when the operating temperature is low, such as for pump impellers located in the cold leg of the reactor coolant system where the temperature is generally at or below 400°C.” where the references are:

Ref. A: T. R. Allen and D. C. Crawford, Lead-Cooled Fast Reactor Systems and the Fuels and Materials Challenges. Science and Technology of Nuclear Installations, Volume 2007, Article ID 97486, doi:10.1155/2007/97486

Ref. B: Vogt, J.-B.; Proriol Serre, I. A Review of the Surface Modifications for Corrosion Mitigation of Steels in Lead and LBE. Coatings 2021, 11, 53. https://doi.org/10.3390/coatings11010053

NRC Response

The staff agrees and has made changes to the “Effect of Flow Velocity” section of the ISG based on this comment. The section now reads:

> “Increasing flow velocities increase the effects of erosion and corrosion and should be considered (IRSN, 2012; Ballinger, and Lim, 2003). Studies have shown that decreasing velocity can control erosion for lead and LBE (Allen T.R. and Crawford D.C., 2007) and for pure lead coolant (Vogt, J.B. and Proriol Serre, I., 2021). Limiting flow velocities are not absolute but temperature- and material-dependent.

> Higher velocities may be acceptable especially for lower operating temperatures. The staff should review temperature, flow velocities and dissolved oxygen to accurately consider the combined effects of erosion and corrosion."

Comment No. 5-24

Section: Page 16, second to last paragraph "Liquid Metal Embrittlement"

Comment/Basis: The ISG says: "The severity of dissolution corrosion attack in Type 316L stainless steel was found to increase with increasing percentages of cold work (Klok et al., 2017)." This is evidence collected in LBE coolant, which is known to behave differently than pure lead.

Recommendation: As such, it is suggested to reword the sentence by saying: "The severity of dissolution corrosion attack in Type 316L stainless steel was found, in LBE coolant, to increase with increasing percentages of cold work (Klok et al., 2017).

NRC Response

The staff agrees with and has incorporated the comment.

Comment No. 5-25

Section: Page 16, "Nonmetallic materials"

Comment/Basis: The ISG says: "SiC/SiC has shown resistance to liquid metal corrosion up to 800 degrees C in a few short-term tests using a non-flowing lead-lithium eutectic (EPRI, 2019b)".

Recommendation: It is recommended not to use this example/reference as it is referred to a coolant, i.e., lead-lithium eutectic, which is not relevant for fission applications and is completely different from the corrosion standpoint with respect to Pb and LBE. The following statement is suggested: "SiC/SiC has shown resistance to liquid metal corrosion up to 550 degrees C in 2000 hr corrosion tests in flowing liquid LBE (Ref. A)". where the reference is:

Ref. A: TAKAHASHI, M., KONDO, M., Corrosion resistance of ceramics SiC and Si3N4 in flowing lead-bismuth eutectic. Progress in Nuclear Energy, Volume 53, Issue 7, September 2011, Pages 1061- 1065. http://refhub.elsevier.com/B978-0-12-803581-8.00749-9/sbref146

NRC Response

The staff agrees the suggested language and reference are appropriate and have incorporated the comment.

Comment No. 5-26

Section: Page 16, bottom of page under "Oxygen control"

Comment/Basis: The ISG says: "Unlike in other reactor types, accelerated corrosion can occur if the dissolved oxygen concentration is either too high or too low at a specific temperature (EPRI, 2019b; Klok et al., 2018)." This statement is not correct since, for example, if the oxygen content is very high and the temperature is below 450-480°C there will not be any significant corrosion (there will be, however, precipitation of lead oxide which should not be confused with corrosion).

Recommendation: In light of this, it is suggested to replace that statement with the following: "Oxygen control is an important technique to ensure satisfactory performance of structural materials in lead- and LBE-cooled reactors. This technique, widely used in lead-based test facilities worldwide (Ref. A), consists of maintaining the oxygen concentration in the coolant within technical specifications, which are generally represented by a minimum required concentration (needed to form a stable passivating oxide layer protecting the underlying bulk material) and a maximum allowed concentration (above which precipitation of lead oxide would occur thus causing plugging concerns for narrow flow passages). The width of this permissible oxygen concentration window is a function of the reactor's operating temperatures (both cold and hot leg temperatures), with the lower end of this window (i.e., minimum required concentration) dependent on the corrosion protection technique leveraged by the class of materials used in the reactor coolant system. For example, adoption of conventional steels such as 316H would rely on the formation of a passivating iron oxide layer which requires a minimum oxygen concentration of approximately 10-8 wt% at 500°C. Relaxation of this limit, thus permitting lower oxygen concentrations thus easing requirements on the oxygen control system, can be achieved by adopting protective coatings/layers either artificially deposited on (or superficially diffused into) the components before they enter service, e.g., aluminization through pack cementation or Al2O3 deposition through Pulsed Laser Deposition, or self-forming/regenerating on the surface of certain materials, e.g., Alumina-Forming Austenitic steels (Ref. B)" where the references are:

Ref. A: M. Tarantino, et al., Overview on Lead-Cooled Fast Reactor Design and Related Technologies Development in ENEA. Energies 2021, 14, 5157. https://doi.org/10.3390/en141651

Ref. B: S. BASSINI et al., "Material Performance in Lead," in Comprehensive Nuclear Materials, Vol. 4, 2nd ed., L.- B. ALLOY, R. J. M. KONINGS, and E. STOLLER ROGER, Eds. pp. 218–241, Elsevier, Oxford (2020).

NRC Response

The staff agrees that the section as originally written could be confusing and has incorporated the comment, in part, to clarify the intent. The ISG now states:

> "Oxygen control is an important technique to ensure satisfactory performance of structural materials in lead- and LBE-cooled reactors. This technique, widely used in lead-based test facilities worldwide (Tarantino M., et al., 2021), consists of maintaining the oxygen concentration in the coolant within controlled limits."

Comment No. 5-27

Section: Advanced Manufacturing Technologies

Comment/Basis: The staff should evaluate whether an application containing AMT components considers (1) the differences between the AMT and traditional manufacturing methods; (2) the safety significance of the identified differences; (3) the aspects of each AMT that are not currently addressed by codes and standards or regulations; and (4) the impacts of the proposed reactor type, operating conditions, and material on the AMT qualification and performance. The staff should confirm that applicants also consider appropriate mitigation strategies, performance monitoring, and surveillance programs to ensure that SSCs fabricated by AMTs continue to satisfy the design criteria.

Recommendation: The guidance should be updated to remove the study of differences between AMT and traditional technologies. Technology is not less safe because it is "newer." Likewise, safety is not affected by technology being "different." Advanced manufacturing Technologies should be evaluated on their own merits and the products of those technological methods. This is most likely what was meant, but clarification is needed.

NRC Response

No changes were made to the ISG based on this comment. While the staff agrees that technology is not inherently less safe because it is newer or that safety is always affected by using a different technology, the staff finds reviewing the differences between traditional manufacturing technologies to be the most effective and expeditious method of reviewing applications.

Letter 6— Anonymous

Comment No. 6-1

In light of the paperwork reduction Act I have prepared these artifacts; namely, to make an example of these criminals who generally place the protection of consumers at risk. </p> State Farm casualty is aoxymoron business…-oxymoron, mostly because they are an automobile insurance company trying to commit crimes in the financial space (not their area of specialization), which is supposed to defraud automobile owners, now also they are defrauding real property taxes and the SEC?? So I also have other questions, like why haven't they been charged for gross negligence of property taxes, the Sarbanes-Oxley Rules…like put their shareholders at risk to hide the Zucker Ponzi scheme?
</p></p>It's the unit dummy, 144 of them valued at what? $20 million in soho??? Good luck with that liability.

NRC Response

The staff made no changes to the ISG based on this comment as it is outside the scope of the ISG.

Letter 7— EPRI

Comment No. 7-1

Page 5/Page 7: Under "General Degradation Mechanisms", degradation mechanisms such as "Corrosion", "Creep and Creep-Fatigue", "Environmentally Assisted Cracking", "Flow Induced Degradation (e.g., Abrasion, Erosion, Cavitation)", "Flow-Induced Vibration", "Irradiation", "Stress Relaxation Cracking" etc. are listed. In the same section, "Gaskets and Seals" is also listed. This sub section is very component specific, rather than a degradation mechanism.

Propose moving the "Gaskets and Seals" discussion to the "General Materials Issues" section.

NRC Response

The staff agrees with and has incorporated this comment.

Comment No. 7-2

Page 6: Editorial suggestion: Below are degradation mechanisms that are likely to apply across different reactor designs, operating environments, and materials. *The degradation mechanisms identified reflect the current state of knowledge. As additional operating experience and laboratory testing become available, the way in which each identified degradation mechanism should be addressed may change. This includes the potential for new degradation mechanisms to be identified.* In the meantime, staff should evaluate whether applicants have adequately addressed the following general degradation mechanisms for various reactor environments.

NRC Response

The staff agrees with and has incorporated this comment, with minor editorial change, "The degradation mechanisms identified reflect the current state of knowledge. As additional operating experience and laboratory testing become available, the way in which each identified degradation mechanism should be addressed may change and new degradation mechanisms may be identified."

Comment No. 7-3

Page 7: As of 2022, ASTM E351 [sic. E531] has been withdrawn.

NRC Response

The staff agrees with this comment and has removed the reference to ASTM D531.

Comment No. 7-4

Page 8: "For example, graphite irradiation strength will increase gradually up to turnaround dose and then will rapidly decrease in strength after turnaround."

In this sentence, "graphite irradiation strength" could be reworded as "strength of irradiated graphite": "For example, strength of irradiated graphite will increase gradually up to turnaround dose and then will rapidly decrease after turnaround."

NRC Response

The staff agrees with and has incorporated this comment.

Comment No. 7-5

Page 9: Editorial suggestion: Below are materials topics that are likely to apply to different reactor designs, coolants, and materials. *The issues identified reflect the current state of knowledge. As additional operating experience and laboratory testing become available, the way in which each identified issue should be addressed may change. This includes the potential for new issues to be identified.* The staff should evaluate whether applicants have adequately addressed the following design neutral materials issues as appropriate for the application and design.

NRC Response

The staff agrees with and has incorporated this comment.

Comment No. 7-6

Page 11: Carburization/decarburization can be an environmental degradation issue for molten salt, Na, and gas-cooled reactors. However, it is only discussed directly in the context of gas-cooled reactors under "He impurities". It is not discussed directly here, for molten salt reactors, but it would be useful to do so.

NRC Response

The staff agrees that Carburization should be considered for all reactor types and has added a Carburization section under General Degradation Mechanisms in addition to a section in each design specific Appendix.

Comment No. 7-7

Page 11: Cladding is only mentioned in “Liquid Metal Reactors” Section as fuel cladding. However, Liquid fuel MSR presents unique challenges for materials selection based on the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC) Section III, Division 5, for use in high-temperature nuclear reactors, because of the high Cr content of code-qualified alloys. In particular MSR concepts, it might be necessary to improve corrosion resistance of already codified alloys to limit wall thinning and eventual loss of structural strength. It could be beneficial to include surface treatments being considered for alloys with less-than-optimal molten salt corrosion resistance, which include weld overlay cladding, electroplating, chemical and physical vapor deposition, hot isostatic pressing, and etc. Good corrosion performance of cladding can be degraded significantly by relatively small alloying additions of other metals. In addition, high temperature microstructural evolution of the coating and the cladding/alloy interface such as changes in grain structure and formation of second phase particles could affect the performance of the cladding. Effective thickness of the cladding is critical since Cr can diffuse through the cladding in MSRs, which could result in formation of voids in the alloy in addition to the Cr loss to the molten salt. Generally, corrosion resistance and mechanical properties under periods of sustained and cyclic loading at high temperatures is needed to assess long-term corrosion protection of cladding. In addition, radiation damage resistance and weldability of the cladding need to be studied.

NRC Response

The staff agrees that cladding could be considered in several reactor designs and has added a section on cladding in the ISG section “Metallic Materials Qualification.”

Comment No. 7-8

Page 13: Carburization/decarburization can be an environmental degradation issue for molten salt, Na, and gas-cooled reactors. However, it is only discussed directly in the context of gas-cooled reactors under "He impurities". It is not discussed directly here, for Na reactors, but it would be useful to do so.

NRC Response

The staff agrees that Carburization should be considered for Na reactors and have added a section on sodium impurities in the ISG.

Comment No. 7-9

Page 15: “Non-code-qualified materials such as alumina forming or aluminum-coated stainless steels and silicon-enriched stainless steels may provide enhanced corrosion resistance in LBE and lead coolants at high temperatures (EPRI,2019b; OECD, 2007; Ballinger and Lim, 2003). The staff should verify that appropriate materials qualification and surveillance programs are in place for any non-code-qualified materials used in lead- or LBE-cooled reactors.”

(1) Does this statement concerning the use of non-code qualified material apply more broadly than just to lead- or LBE-cooled reactors?
(2) One assumes that there is a minimum level of "qualification" in order to activate this approach. Therefore, can "appropriate qualification" be clarified?

NRC Response

The staff have removed this section of the ISG. See response to Comment 7-10.

Comment No. 7-10

Page 15: General comment pertaining to the use of "non-code-qualified materials"

This seems inconsistent with the rest of the document, e.g.:

Page 1: “The guidance in this document identifies areas of staff review that could be necessary for a submittal seeking to use materials allowed under American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME Code), Section III, “Rules for the Construction of Nuclear Facility Components,” Division 5, “High Temperature Reactors” (Section III-5) (ASME, 2017).”

Page 19: IMPLEMENTATION. This section references that this ISG will be used to review non-LWR applications that propose to use materials allowed under ASME Section III, Division 5.

It is proposed that the statement concerning the use of "non-code-qualified materials" be located in a section applicable to all reactor designs, not solely under the verbiage on page 15.

NRC Response

The staff removed the cited paragraph and has added generic guidance on non-code qualified materials in the Applicability section of the ISG.

Comment No. 7-11

Page 19: “The staff should evaluate whether there is a favorable environment that leads to a stable oxide film and stable internal carbides (INL, 2006) and avoids excessive carburization, surface carburization, and decarburization. Other environmental factors to evaluate are the effects of temperature, alloy composition, and *other impurities such as H2O* (NRC, 2021c).”

Perhaps use another impurity as an example, as H2O was discussed in the previous paragraph.

NRC Response

The staff agrees that the example of H2O was an unnecessary duplication and made the following modification to the ISG: “Other environmental factors to evaluate are the effects of temperature, alloy composition, and other coolant impurities.”

Comment No. 7-12

Page 19: The reference for Figure 7 should be NRC 2021c, not NRC 2021a.

NRC Response

The staff agrees with and has incorporated this comment.

Comment No. 7-13

Page 19: Propose statement under “Metallic Materials Qualification” be moved under the "General Materials Issues" section, as the qualification of metallic materials will be pertinent for designs beyond HTGRs.

NRC Response

The staff agrees with this comment that qualification of metallic materials will be pertinent for designs beyond HTRGs and moved the section to the General Materials Issues section of the ISG.

Letter 8 — INL

Comment No. 8-1

We strongly support the emphasis on performance monitoring and surveillance specimens/testing to address materials degradations during reactor operations.

We note that the need for performance monitoring and surveillance programs, particularly for very long design lifetimes, e.g., 500,000 hours, has been reinforced in Division 5 recently. A General Note has been added to Table HBB-I-14.10E-1 on the stress rupture factors for 9Cr-1Mo-V weldment in the 2023 edition of Division 5 which states:

> *The values in this table are extrapolated from shorter term test data using an engineering model. For longer design lives, the designer should consider further strength reductions to account for potential in-service material degradation, per HBB-2160(a). In addition, enhanced material surveillance programs and/or heightened in-service inspection per the rules of ASME Section XI may be warranted.*

NRC Response

The staff agrees with this comment but found no edits were needed to the ISG based on this comment.

Comment No. 8-2

Since some Advanced Non-Light Water Reactors (ANLWRs) may be used in whole or in part for the generation of nuclear process heat that will be used in an associated facility (e.g., hydrogen generation, ammonia production, petrochemical refining, etc.), it will be very important to provide guidance as to where the nuclear island stops and the non-nuclear facility begins with regard to safety standards and design margins for the structures, systems, and components (SSCs). This is particularly important for secondary or tertiary heat transfer loops. Also, the potential for adverse feedback between the nuclear and non-nuclear portions of a site--going either way--must be considered.

NRC Response

The staff agrees with this comment; however, it is out of scope of the ISG, and no changes were made to the ISG based on this comment.

Comment No. 8-3

How will the safety significance and potential consequences of SSC malfunctions or failures be assessed to set required margins and assess design adequacy? Considering that one goal of ANLWRs is to produce power more economically, vendors will likely consider the use of reduced margins or commercial design codes where possible/appropriate. Augmented staff guidance to evaluate the adequacy of these approaches should be provided.

NRC Response

The staff made no changes based on this comment as it is outside the scope of the ISG.

Comment No. 8-4

It is agreed that appropriate mitigation strategies, performance monitoring, and surveillance programs should be considered to address the effects of thermal aging on design properties (ISG p8). Though, it is noted that thermal aging effects on yield and ultimate strength are specifically addressed in Division 5 and mandatory factors are provided.

NRC Response

The staff agrees with this comment and determined no changes were necessary to be made to the ISG.

Comment No. 8-5

While components fabricated with advanced manufacturing technology (AMT) are addressed in the ISG (pp 9-10), there does not appear to be guidance regarding assuring that their high temperature properties are adequately defined. While this is an emerging field, limited studies have shown that while some AMT materials may have comparable room temperature or even short-term elevated temperature properties, long-term creep, fatigue, and creep-fatigue properties may be significantly reduced from wrought material values. Highlighting this issue and providing guidance or references related to it would be valuable. The ASME Section III Task Group on Division 5 AM Components would be a good source for up-to-date information on this subject.

NRC Response

The staff agrees that staff should consider potential performance differences of AMT fabricated components operated at high temperatures and has incorporated the comment under General Materials Issues section “Advanced Manufacturing Technologies”. The section now includes this statement, “It is particularly important that an application fully addresses AMT material performance at high temperatures. Limited studies have shown long-term creep, fatigue, and creep-fatigue properties may be reduced compared to wrought material values”.

Comment No. 8-6

While wear and fretting are mentioned in the ISG (pg. 9), it is important to note that tribology is significantly affected by particular coolants. Limited results have shown issues related to self-welding of SSCs in helium coolants. The virtual elimination of external oxide layers on metallic components in fluoride salt coolants practically ensures different tribological behavior in such media. Staff need to be given guidance to assess tribology in specific reactor environments.

NRC Response

The staff agrees that tribology is an important consideration and incorporated this statement into “Coolant Flow, wear and fretting” in the ISG. The section now includes this statement “Reactor operation may be affected by tribological effects such as friction, wear and fretting”.

Comment No. 8-7

Guidance on ensuring that irradiation effects are adequately addressed for ANLWRs is described in multiple places in the ISG. However, given that the high doses likely to be reached in some fast reactor components can significantly exceed the existing data base on irradiation effects, it could be valuable to provide additional guidance regarding the potential value and limitations for using high-dose, ion-beam irradiations to further assess property changes at these high doses. The recent work funded as part of DOE's Integrated Research Program and lead by the University of Michigan on high-dose ion-irradiation effects probably provides the most comprehensive information on this subject currently available.

NRC Response

The staff acknowledges the work done under the DOE Integrated Research Program and the University of Michigan. However, the staff has not evaluated whether the use of ion-beam irradiation as a substitute for neutron irradiation effects on different structural materials is acceptable. Therefore, no changes were made to this ISG.

DANU-ISG-2023-01 APPENDIX B

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