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Material Compatibility for non-Light Water Reactors

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DRAFT INTERIM STAFF GUIDANCE

MATERIAL COMPATIBILITY FOR NON-LIGHT WATER REACTORS

DANU-ISG-2023-01

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-significant 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 of component, 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; however, the staff does not currently have guidance on how it should review such programs. Such guidance on appropriate qualification, performance monitoring methods and in-service inspection will be useful for the NRC staff in reviewing 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 also present different materials challenges that have not been encountered previously. 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 the NRC staff would need to review depends on many factors, including the maturity of the AMT process in codes and standards, applicable precedents, and 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 environmental challenges to material performance that are not present in light water reactors (LWRs) as the operating environments for non-LWRs are different than those in the current fleet of LWRs. One consideration is that the 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 the 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, there are many non-LWR designs with a variety of coolants, which create unique operating environments for reactor materials and components. This ISG provides non-plant-specific guidance in the discussion section below. In addition, Parts 1, 2 and 3 provide technology-specific guidance for molten salt reactors (MSRs), liquid metal reactors, and high-temperature gas-cooled reactors, respectively.

APPLICABILITY

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

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

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 principal design criteria for non-LWR designs. The RG also describes the NRC's proposed guidance for modifying and supplementing the GDC 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 leaktight 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.

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 will 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 appropriate QA programs were followed when reviewing a materials qualification program[2].

[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

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 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.

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. 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 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

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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).

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

Corrosion

The staff should ensure that corrosion is assessed as a function of temperature, time, microstructure, coolant composition and (as appropriate) 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 consider appropriate mitigation strategies, performance monitoring, and surveillance programs to ensure that SSCs affected by corrosion continue to satisfy the design criteria.

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 code, 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 is assessed as a function of temperature, time, microstructure, coolant composition and (as appropriate) chemistry and coolant flow conditions. 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 consider appropriate mitigation, performance monitoring, and surveillance programs to ensure that SSCs affected by flow-induced vibration continue to satisfy the design criteria.

Gaskets and Seals

The staff should verify 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 also consider appropriate mitigation, performance monitoring, and surveillance programs to ensure that gaskets/seals in contact with coolant 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, 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 dimension or mechanical properties so that they no longer meet their design function. 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. The staff should consider the most recent version of American Society for Testing and Materials (ASTM) E531, “Standard Practice for Surveillance Testing of High Temperature Nuclear Component Materials,” when reviewing surveillance testing.

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 value will change to decreasing value) with increasing received neutron dose. This critical dose level where the change occurs has been termed the "turnaround dose" and indicates when the graphite material irradiation induced dimensional volumetric densification reverses to a volumetric expansion behavior. For example, graphite irradiation strength will increase gradually up to turnaround dose and then will rapidly decrease in strength after turnaround. The exception to this irradiated material behavior is thermal diffusivity which experiences an immediate and significant decrease followed by a gradual decrease in value with increasing accumulated dose.

Stress Relaxation Cracking

The staff should ensure that the potential for stress relaxation cracking (SRC) is assessed. RG 1.87 states that 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 licensing period 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 also 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 stripping 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 synergistic effects of thermal fatigue, vibratory fatigue, and creep-fatigue are addressed by the applicant.

Wear/Fretting

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.

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 different reactor designs, coolants, and materials. The issues identified reflect the current state of knowledge; however, as additional operating experience and laboratory testing become available, the need to address how each identified issue 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 the 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. 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.

Ceramic Insulation

The staff should evaluate whether an application adequately addresses environmental effects on ceramic insulation. For example, the effects of irradiation and thermal stability on the structural integrity of fibrous ceramic insulation, chemical compatibility of ceramic insulation with the coolant, and the potential for off-gassing from ceramic insulation under operational conditions may affect both the performance of sensors located near the insulation and the potential for oxidation under accident conditions. The staff should confirm that applicants also consider appropriate mitigation strategies, performance monitoring, and surveillance programs to ensure that ceramic insulation continues to satisfy the design criteria.

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.

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). Oil-based lubricants should not be used in components interfacing with the coolant.

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 consider the response of monolithic SiC, C/C, and SiC/SiC to irradiation in its review.

The staff should be aware that nonmetallic composites have the potential for use in HTR 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 draft ISG (ASME, 2021). There is no operational experience with this class of materials in experimental, prototype, or commercial reactors.

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 staff should verify that each grade of SiC/SiC is qualified separately. 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.

SA-508/533 Bainitic Steel for Reactor Pressure Vessels

The staff should evaluate the potential for nonhardening embrittlement caused by thermal aging in the coarse-grained, heat-affected zone weldments of SA-508/533 steels (EPRI, 2000; Nanstad et al., 2018). In addition, the staff should consider creep effects for the use of SA-508/533 steels at temperatures above 371 degrees Celsius (C) in the reactor pressure vessel for a lifetime of 60 years or greater (EPRI, 2020a). The staff should verify that applicants also consider appropriate mitigation strategies, performance monitoring, and surveillance programs to ensure that the reactor pressure vessel continues 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 molten salt coolant has relatively small amounts of fissile material and fission products. The fissile material and fission products are contained within a TRISO (tristructural isotropic particle fuel) fuel particle, which could be in a prismatic graphite compact or pyrolytic graphite sphere. 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 (ORNL, 2021a). The staff should confirm that graphite monitoring programs reflect any compatibility issues identified.

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).

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 wear, abrasion, and/or erosion from the dense molten salt coolant on the softer graphite and carbon-carbon composites.

The staff should verify that applicants also consider 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 consider 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 considers 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 consider 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). The sodium-cooled fast reactor (SFR) has had decades of experience at the experimental, prototype, and commercial scale. 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 propulsion nuclear reactors in Alfa-class submarines operated by the Soviet Union from 1967–1983 (EPRI, 2019b; IRSN, 2012); 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 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 ingress of moisture and therefore to caustic cracking in sodium (NRC, 2019). A significant finding related to SFRs is that austenitic stainless steels are unsuitable for SFR steam generators because of the potential for caustic SCC following even small leaks (NRC, 2019). 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 consider 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 this phenomenon will not occur. 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 consider appropriate mitigation strategies, performance monitoring, and surveillance programs to minimize the potential for contact between molten sodium and water or moisture in the air.

Galvanic Corrosion

The staff should evaluate whether the application adequately considers other environmental factors in addition to sodium purity, for example, proximity to potentially carburizing materials, such as boron carbide (B_4C) fuel rod elements (NRC, 2021e). The staff should consider the potential for accelerated corrosion if dissimilar materials are used in the same sodium coolant loop (NRC, 2021e). The staff should verify that systems are designed to minimize the potential for corrosion or material degradation by dissimilar materials, or that the applicant implements appropriate mitigation, monitoring, and surveillance programs to manage such effects.

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 below 1 part per million (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 (flow rate and impurities, particularly 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.

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 is a strong function of temperature (Ballinger and Lim, 2003; EPRI, 2019b). As a result, use of typical ferritic and austenitic steels requires special treatments, such as alloying additions or coatings (EPRI, 2019b). Specific data of the environmental impacts of molten lead and LBE on materials are not interchangeable.

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). Other currently available materials, such as 9Cr-1Mo-V and 15-5Ti stainless steel, also suffer more rapid corrosion at temperatures greater than or equal to 550 degrees C in a lead or LBE environment (EPRI, 2019b).

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.

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

Lead is highly eroding, and for this reason, the flow velocity should be limited (IRSN, 2012; Ballinger and Lim, 2003). The staff should verify that applicants have considered the effect of coolant flow velocity on corrosion of structural materials.

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 to increase with increasing percentages of cold work (Klok et al., 2017).

The staff should verify that the applicant considers 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 800 degrees C in a few short-term tests using a non-flowing lead-lithium eutectic (EPRI, 2019b). 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. 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). Corrosion rates at temperatures below 450 degrees C 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 consider 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 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. The NRC is not aware of any current plans to deploy GFR reactors in the United States, so this section does not address materials concerns for GFRs. Common features of these designs include a reactor outlet temperature greater than or equal to 700 degrees C. 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 consider 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 consider 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 consider 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 such as H_2O (NRC, 2021c). Figure 7 in NRC 2021a 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).

Metallic Materials Qualification

The staff should verify that metallic materials to be used in structural components in HTGRs have been qualified for use in a representative helium 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, and decarburization resulting from the helium coolant environment.

IMPLEMENTATION

The NRC 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

Discussion to be provided in the final ISG.

CONGRESSIONAL REVIEW ACT

Discussion to be provided in the final ISG.

FINAL RESOLUTION

The NRC 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 Reserved
B References

APPENDIX A

Reserved

APPENDIX B

References

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