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Material Compatibility for Non-Light-Water Reactors

Draft Interim Staff Guidance

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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 non-light-water reactor (non-LWR) designs, including both power and nonpower 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 (BPVC), Section III, “Rules for the Construction of Nuclear Facility Components,” Division 5, “High Temperature Reactors” (III-5) (ASME, 2023), or under other commercial codes and standards used for the design and construction of mechanical components.¹ The standards in III-5 specify the mechanical properties and allowable stresses to be used for design of components in high-temperature reactors (HTRs). However, as stated in III-5, HBB-1110(g), the ASME BPVC 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 when reviewing applicable design requirements, including materials performance; environmental compatibility; 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 materials performance and monitoring programs depends on many factors, such as plant design, safety importance of SSCs, specific environments, and maturity of research in a given area. The staff should consider these factors 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”; 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”; and non-LWR applications for standard design approvals, standard design certifications, manufacturing licenses, construction permits, operating licenses, and combined licenses under 10 CFR Part 53, “Risk-Informed, Technology-Inclusive Regulatory Framework for Commercial Nuclear Plants.”

Appendix A to this ISG, “Content of Applications for Materials Performance for Advanced Non-Light-Water Reactors,” describes information the NRC staff generally has found necessary to perform its review of materials of construction and passive SSC performance for non-LWRs. As noted above, the information necessary for reviewing the application will depend on many

¹ Similarly, the NRC staff endorsed, with conditions, the 2019 edition of ASME BPVC, Section XI, Division 2, “Requirements for Reliability and Integrity Management (RIM) Programs for Nuclear Power Plants” (XI-2) for use by non-LRW applicants and licensees subject to 10 CFR Parts 50 and 52 in RG 1.246, Revision 0 (NRC, 2022b). The NRC staff has not endorsed either of these documents for use by applicants and licensees subject to 10 CFR Part 53. Both documents are referenced throughout this ISG. The NRC staff is currently considering whether to endorse both III-5 and XI-2 for applicants and licensees subject to 10 CFR Part 53. Those references do not mean III-5 and XI-2 are currently acceptable for use by applicants and licensees under 10 CFR Parts 53. If, based on the NRC staff’s review of both codes, the NRC staff determines that it cannot endorse either III-5 or XI-2 for use by applicants and licensees subject to 10 CFR Part 53, it will revise this ISG accordingly before issuing it for use.

factors. The appendix incorporates lessons learned by the NRC staff from recent non-LWR licensing reviews. It is also informed by preapplication engagement, topical reports, and information that has historically been supplied for LWR applications. The appendix is intended to provide guidance on the content of applications in this area, to enable efficient staff reviews as well as regulatory consistency and predictability.

BACKGROUND

In its review of non-LWR applications, the NRC evaluates whether structural materials will allow components to fulfill design requirements over the intended design life, and whether adequate surveillance and monitoring programs are in place. Regulations in 10 CFR Parts 50, 52, and 53 include requirements for assuring adequate materials performance and performance monitoring. The NRC staff has identified the need for guidance on materials performance and performance monitoring to support staff reviews of applications for construction permits or operating licenses under 10 CFR Parts 50 or 53, or for design certifications, combined licenses, standard design approvals, or manufacturing licenses under 10 CFR Parts 52 or 53, that propose to use materials allowed under III-5 or other commercial codes and standards.

New fabrication methods present new material considerations for staff reviews. In place of components manufactured by conventional processes (e.g., forging, castings), an applicant may propose to use components fabricated with advanced manufacturing technologies (AMTs), such as laser powder bed fusion or directed energy deposition additive manufacturing. The materials produced by these technologies may differ from conventionally produced materials in their microstructures, or in the types of defects they are prone to. Post-fabrication processing requirements may also differ. Therefore, appropriate manufacturing controls should be applied to ensure that the material properties of the components produced are acceptable and are confirmed through proper testing. The information related to AMTs that the staff would need to review for a given application 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 each component. The NRC has developed generic draft guidelines (NRC, 2021a) and AMT-specific draft guidelines (e.g., NRC, 2021b) for considering the following elements of a submittal that may use AMT components: quality assurance, AMT process qualification, supplemental qualification testing, production process control and verification, and performance monitoring. As noted in these documents, they do not represent official agency positions.

Because non-LWRs differ from currently operating light-water reactors (LWRs) in their operating temperatures and coolant types, they present novel environmental challenges to materials performance. Non-LWRs may operate at significantly higher temperatures than current nuclear power plants; in particular, their operating temperatures may correspond to the creep regime in which plastic deformation may occur at applied stress below the yield stress. To address this possibility, Regulatory Guide (RG) 1.87, Revision 3, "Acceptability of ASME Code, Section III, Division 5, 'High Temperature Reactors,'" issued August 2025 (NRC, 2025a), endorses the use of III-5, with conditions. III-5 considers mechanical and thermal stresses due to cyclic operation and high-temperature creep; 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.²

² See III-5, paragraph HAA-1130, "Limits of These Rules."

Non-LWRs also use different coolants from 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. Such coolant environments may increase the susceptibility of components to material corrosion, various degradation mechanisms, and irradiation effects. Numerous studies have identified the gaps in knowledge that exist for certain coolant types, as well as the impact of the use of such coolants on the materials being considered for use in the construction and operation of non-LWR plants (NRC, 2003; INL, 2006; ANL, 2017; EPRI, 2019a; EPRI, 2019b; EPRI, 2020a; EPRI, 2020b; NRC, 2021c; NRC, 2021d; NRC, 2021e; NRC, 2024a).

This ISG provides NRC staff guidance on the review of materials areas not covered by III-5, such as the effects of corrosion, radiation, or other environmental conditions. It identifies information the staff should consider in its review of environmental compatibility and materials performance. It also indicates where it may be appropriate for licensees to rely on monitoring and surveillance to ensure component integrity.

As noted above, the coolants proposed for various non-LWR designs create unique operating environments for reactor materials and components. The section “Discussion” below gives non-plant-specific guidance for non-LWRs. The subsequent sections provide technology-specific guidance for molten salt reactors (MSRs), liquid metal reactors, and high-temperature gas-cooled reactors.

APPLICABILITY

This ISG is applicable to NRC staff reviews of applications for non-LWR designs, including power and nonpower reactors, for permits, licenses, certifications, and approvals under 10 CFR Parts 50, 52, or 53. 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.

The contents of this ISG were developed with a focus on ensuring materials performance for components designed to III-5 by providing guidance on topics related to those materials but not covered by III-5. However, whether an applicant is proposing to use III-5 or another code or standard, this guidance is applicable and the staff should make sure the application addresses the considerations in this ISG related to the compatibility of these materials with the corresponding environments.

GUIDANCE

Current Regulatory Framework

10 CFR Parts 50 and 52

Under 10 CFR 50.34(a)(3)(i), 10 CFR 52.47(a)(3)(i), and 10 CFR 52.79(a)(4)(i), the applicant must include principal design criteria (PDC) for the facility. As described in Appendix A to 10 CFR Part 50, the principal design criteria establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety; that is, structures, systems, and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. The General Design Criteria (GDC) in Appendix A to 10 CFR Part 50 establish minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in

design and location to plants for which construction permits have previously been issued by the Commission. However, Appendix A to 10 CFR Part 50 is also considered generally applicable to other types of nuclear power units and provides guidance in establishing PDC 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 guidance for the development of PDC for non-LWR designs. The RG also provides guidance for modifying and supplementing the general design criteria to develop PDC appropriate to two specific non-LWR design concepts: sodium-cooled fast reactors and modular high-temperature gas-cooled reactors. The following criteria are related to structural materials compatibility and performance for these reactors:

- Advanced Reactor Design Criterion (ARDC) 1, Sodium Fast Reactor Design Criterion (SFR-DC) 1, and Modular High-Temperature Gas-Cooled Reactor Design Criterion (MHTGR-DC) 1 state, in part, that SSCs important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions being performed.
- ARDC 4, SFR-DC 4, and MHTGR-DC 4 state, 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, SFR-DC 14, and MHTGR-DC 14 state that the reactor coolant boundary, the primary coolant boundary, and the helium pressure boundary, respectively, shall be designed, fabricated, erected, and tested so as to have an extremely low probability of, among other things, abnormal leakage, of rapidly propagating failure, and of gross rupture.
- ARDC 30, SFR-DC 30, and MHTGR-DC 30 state, in part, that components that are part of the reactor coolant boundary, the primary coolant boundary, and the reactor helium pressure boundary, respectively, shall be designed, fabricated, erected, and tested to the highest quality standards practical.
- ARDC 31, SFR-DC 31, and MHTGR-DC 31 state, in part, that when stressed under operating, maintenance, testing, and postulated accident conditions, (1) the reactor coolant boundary, the primary coolant boundary, and the reactor helium pressure boundary, respectively, behave in a nonbrittle manner, and (2) the probability of rapidly propagating fracture is minimized.
- ARDC 32, SFR-DC 32, and MHTGR-DC 32 state that components that are part of the reactor coolant boundary, primary coolant boundary, and reactor helium pressure boundary, respectively, shall be designed to permit (1) periodic inspection and functional testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor vessel. Similar ARDC, SFR-DC, and MHTGR-DC exist requiring that other systems performing heat removal and containment functions be designed to permit appropriate periodic inspection.
- SFR-DC 71 states, in part, that systems shall be provided as necessary 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 inspection and functional testing of important areas and features to assess their structural and leaktight 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 ARDC and some SFR-DC will likely apply to MSRs. MSR applicants may also propose additional design criteria not discussed in RG 1.232. Furthermore, while RG 1.232 does not explicitly consider non-LWR nonpower reactors, the design criteria listed above may be used to inform the development of PDC related to structural materials performance in such reactors.

The staff should confirm that the application gives sufficient information on materials performance, mitigation strategies, performance monitoring, and surveillance programs to demonstrate that established facility-specific PDC are sufficiently satisfied based on the type of application being reviewed.

10 CFR Part 53

In contrast to describing prescriptive design rules and requirements, such as the GDC under Appendix A to 10 CFR Part 50, 10 CFR 53.410 and 53.420 requires applicants to establish functional design criteria (FDC) for each design feature proposed in an application. This risk-informed and performance-based approach in Part 53 allows applicants flexibility to propose design features to meet safety objectives and achieve safety outcomes. The term FDC is defined in 10 CFR 53.020 as metrics for the performance of structures, systems, and components (SSCs). Under 10 CFR 53.020, for safety-related SSCs, FDC define performance metrics necessary to demonstrate compliance with the safety criteria in § 53.210. Under 10 CFR 53.020, for non-safety-related, but safety significant SSCs, FDC define performance metrics necessary to demonstrate compliance with the safety criteria in § 53.220. In that regard, the FDC are part of a set of requirements under Part 53 where high-level safety criteria are met through achieving safety functions; safety functions are achieved through the performance of design features relied upon to demonstrate that the safety criteria are met; and the design features must meet the performance metrics defined by their respective functional design criteria.

Under 10 CFR 53.440, analysis, appropriate test programs, prototype testing, operating experience, or a combination thereof must demonstrate how the performance of each safety design feature will be capable of fulfilling FDC considering interdependent effects.

In addition, 10 CFR Part 53 contains specific requirements related to passive SSC performance. Under 10 CFR 53.440(b), design features classified as SR must, wherever applicable, be designed using generally accepted consensus codes and standards that have been endorsed or otherwise found acceptable by NRC. 10 CFR 53.440(c) requires SR and NSRSS SSCs to use materials qualified for their service conditions over the design life of the SSC as appropriate to satisfy the special treatments established for the SSC under 10 CFR 53.460. Finally, under 10 CFR 53.870, each holder of an OL or COL must develop, and implement, and maintain an integrity assessment program to monitor, evaluate, and manage material degradation and aging. Under 10 CFR 53.440(d), possible degradation mechanisms related to aging, fatigue, chemical interactions, operating temperatures, effects of irradiation, and other environmental factors that may affect the performance of SR and NSRSS SSCs must be evaluated and used to inform the design and the development of integrity assessment programs under 10 CFR 53.870. These specific requirements along with the defined FDC will generally cover similar aspects of structural materials compatibility and performance as the PDC under 10 CFR Parts 50 and 52 discussed in the prior section.

Other Relevant Guidance

The following existing guidance documents pertain to materials performance and are directly related to the contents of this ISG:

- RG 1.87, Revision 3 (NRC, 2025a) describes an approach that is acceptable to the NRC staff for ensuring the mechanical/structural integrity of components operating in elevated-temperature environments subject to time-dependent material properties and failure modes by endorsing, with exceptions and limitations, the 2023 edition of III-5 and several related code cases. RG 1.87, Revision 3 applies to non-LWR licensees and applicants subject to 10 CFR parts 50 and 52.
- RG 1.246, Revision 0 “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,” (NRC, 2022b) describes an approach that is acceptable to the NRC staff for the development and implementation of preservice inspection and in-service inspection programs for non-LWRs by endorsing, with conditions, the 2019 edition of ASME BPVC, Section XI, Division 2, “Requirements for Reliability and Integrity Management (RIM) Programs for Nuclear Power Plants” (XI-2). RG 1.246 applies to nuclear power reactor applicants and licensees for combined or operating licenses for reactors with non-LWR designs subject to 10 CFR parts 50 and 52.
- DANU-ISG-2022-07, “Advanced Reactor Content of Application Project: Risk-Informed In-service Inspection/In-service Testing Programs for Non-LWRs,” (NRC, 2023) provides guidance on the contents of applications associated with risk-informed in-service inspection and in-service testing programs to an applicant submitting a risk-informed, performance-based application for a construction permit or operating license under 10 CFR Part 50 or a combined license, manufacturing license, standard design approval, or design certification under 10 CFR Part 52 for a non-LWR. The ISG also provides guidance to the NRC staff on how to review such an application. Although

DANU ISG-2022-07 is not applicable to 10 CFR Part 53, it may provide useful insights for applications under Part 53.

Discussion

Environmental Compatibility Assessment and Performance Monitoring

This ISG addresses important aspects of an environmental compatibility assessment to assure that potential degradation mechanisms are appropriately identified and managed. An SSC's performance should be demonstrated through a combination of new and existing test data, analysis, and performance monitoring, which collectively provide assurance that the SSC will meet the design requirements over its intended design life in the applicable operating environment. Appendix A to this ISG identifies information that the staff should consider during its review of materials performance for non-LWR applications.

Quality assurance (QA) is a process followed to ensure that a component meets 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. They may also establish requirements for supplemental testing, performance monitoring, and surveillance programs. The staff should confirm that an appropriate QA program was followed when reviewing the basis for assuring materials performance.³

The selection of structural materials for the reactor design should consider how the operating environment will affect materials properties and allowable stresses. Environmental compatibility assessments should consider experimental test data (including historical or legacy data) collected in an environment sufficiently representative of the anticipated operating environment for the SSC, including chemistry, temperature, and irradiation. These assessments should consider how applicable the data are to the plant design and environment and to account for uncertainties in the environment, material composition, fabrication methods, and operating conditions. Environmental compatibility assessments should cover materials for safety-related components, safety-significant components, and, as needed, non-safety-related components whose failure could impact important safety functions. Appendix A to this ISG provides additional guidance on the scope of review of materials performance for non-LWR applications.

When there is uncertainty about whether degradation will occur or the extent to which it will progress during reactor operation, performance monitoring may be used to provide additional assurance that SSCs will continue to meet design requirements. Elements of performance monitoring may include periodic inspections, continuous monitoring, or surveillance coupon activities, the need for which would be determined by the reactor design and operational model. For example, data gathered from surveillance coupons can be particularly useful for assessing changes in material properties, such as embrittlement due to irradiation or high temperature exposure, that can then be used to benchmark models for predicting the degradation of components within the reactor. For materials used in components for which the knowledge base to predict design life is limited, performance monitoring and surveillance programs may be an

³ While a QA program description is not required to be submitted or approved as part of a nonpower reactor operating license application, as part of its review of such an application, the staff will determine whether the 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 ensure safe operation, as required by 10 CFR 50.34(b)(6)(ii).

effective way to ensure materials and component performance. A component with a significant design margin or one that has demonstrated acceptable performance in similar operating environments and conditions may require less rigorous performance monitoring. When determining whether a particular design is acceptable, the staff should holistically consider all relevant factors, including susceptibility to degradation, design margin, prior operating experience, consequences of failure, defense-in-depth, and performance monitoring. XI-2 provides one method for developing a comprehensive aging management strategy, subject to NRC acceptance of the proposed program. Appendix A to this ISG provides additional guidance on how to evaluate performance monitoring approaches to provide assurance of materials performance.

General Degradation Mechanisms

This section describes several degradation mechanisms that are likely to apply across different reactor designs, operating environments, and materials. The information below reflects the current state of knowledge. As additional operating experience and laboratory testing data become available, the way these mechanisms should be addressed may change, and new mechanisms may be identified. In the meantime, the staff should evaluate whether applicants have adequately addressed these degradation mechanisms for the relevant reactor environments.

Carburization and Decarburization

Carburization and decarburization of structural alloys can increase degradation rates of these materials (Chan et al., 2018; NRC, 2003; NRC, 2021d; NRC, 2021e; Sridharan, 2019). As noted in the design-specific sections below, carburization and decarburization have different causes depending on the environment and other materials present in the design. The staff should review the interactions of graphite and carbon impurities in the coolant with metals to ensure that qualification testing, monitoring, surveillance, and inspection programs adequately address potential carburization and decarburization.

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. Localized corrosion, galvanic effects, leaching, erosion/wear, and coolant solubility-driven corrosion effects should also be considered. The staff should confirm that the applicant has 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 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 the applicant has considered 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 the applicant has assessed environmentally assisted cracking mechanisms, including stress-corrosion cracking (SCC), intergranular cracking, and fatigue cracking. Based on operating experience and laboratory studies conducted in LWRs, environmentally assisted cracking is most likely to be significant in weld metal or in the heat-affected zone. It is important that component design minimize the potential for crack initiation, and that there be sufficient flaw tolerance to fabrication and service cracking. The staff should verify that the applicant has considered 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 leading to loss of material and/or wall thinning 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 flowpath, affecting coolant flow patterns and local heat transfer properties. The staff should also verify that pumps are qualified and tested under operating conditions and coolant flowpaths, and that flow rates are evaluated to minimize the potential for cavitation. For graphite components, the staff should evaluate whether the application adequately addresses the potential for wear, abrasion, or erosion between graphite components; the generation of dust and debris; and graphite component geometry. The staff should confirm that the applicant has considered 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 vibrations induced by coolant flow, which may cause fretting and fretting-assisted fatigue. In addition, the staff should confirm that the frequency of flow-induced excitations is not close to the natural frequency of the system. The staff should confirm that the applicant has considered appropriate mitigation strategies, 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. Irradiation can change component dimensions and mechanical properties such that they no longer meet their design functions. Hence 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 the applicant has considered appropriate performance monitoring and surveillance programs to ensure that SSCs affected by irradiation continue to satisfy the design criteria. It may be appropriate to supplement a materials qualification program with the use of test specimens within the reactor (e.g., coupon

specimens irradiated during reactor operations) that can be withdrawn and tested throughout the operating phase of the reactor — for example, to support longer lifetimes or supplement areas with minimal existing data.

RG 1.87, Rev. 3, provides guidance related to the irradiation and oxidation of graphite. Above temperatures of about 300 degrees Celsius (°C), fast neutron irradiation initially causes graphite to undergo a volumetric contraction until it reaches a critical fluence known as “turnaround,” beyond which it begins to expand. As fluence increases, the graphite returns to its initial volume at the “crossover” fluence, after which it continues to expand beyond its initial volume. In general, the strength and elastic modulus of graphite will increase with increasing neutron irradiation until the turnaround fluence is reached, upon which they will begin to decrease. Generally, irradiated graphite will have greater strength and modulus than unirradiated graphite until the crossover fluence is reached.

There is limited experimental data on graphite irradiated below 300°C, but some data have shown that the graphite at these temperatures does not undergo an initial volumetric contraction followed by expansion as it does at higher temperatures. Instead, it expands immediately (i.e., there is no turnaround fluence). Given the implications of this behavior on graphite performance, the staff should closely review the justification for any thermomechanical models or changes to graphite behavior for graphite irradiated below 300°C.

In addition, at lower irradiation temperatures, Wigner energy (potential energy due to irradiation damage) can build up in graphite components. If the temperature of the graphite is then increased relative to the temperature during irradiation, the Wigner energy can be released, producing heat and substantially increasing the temperature of the graphite. III-5 states that designers should account for Wigner energy in graphite components irradiated below 200°C. This temperature approximates a transition point where atomic mobility anneals out a significant fraction of defects (ORNL, 2011). Between 200°C and 300°C, Wigner energy can build up under high neutron flux and should also be considered. Above 300°C, Wigner effects can generally be considered insignificant under most conditions. The staff should evaluate the potential for Wigner energy buildup based on the neutron flux and other relevant parameters, as well as the potential safety implications of a release of Wigner energy during a transient.

Stress Relaxation Cracking

The staff should ensure that the potential for stress relaxation cracking (SRC) is assessed. Consistent with RG 1.87, Rev. 3, applicants should submit a plan for addressing SRC. SRC has been studied extensively over a long period of time. It is also sometimes called reheat cracking, stress relief cracking, strain-age cracking, ductility dip cracking, or postweld heat treatment 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 promote SRC, including, but not limited to, weld residual stresses, cold work, larger grain sizes, multiaxial stresses, notches, and constraints caused by the weld joint design. SRC occurs in austenitic alloys within alloy-specific temperature ranges (Colwell and Shargay, 2020; Shoemaker et al., 2007; van Wortel, 2007; Miller, 1998; API, 2017; NRC, 2019; ASME, 2020; ASME, 2021; NRC, 2024a). Ways 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 the applicant will implement appropriate preventive measures during design, construction, and operation, including potential post-startup weld inspections and monitoring.

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 the applicant has considered 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). Greater surface roughness generally increases emissivity but has a negative effect on fatigue performance. The staff should closely review designs that increase surface roughness to increase emissivity to ensure that impacts on fatigue performance are appropriately addressed and the required emissivity values will be achieved. The staff should also review other surface treatment methods to increase emissivity including thermally grown oxide or carbide layers.

If the reactor design specifications rely on thermal emissivity (e.g., for heat rejection), then the staff should confirm that the applicant has considered how exposure to the coolant, irradiation, or ambient air at elevated temperatures could affect the emissivity of materials. 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 the application adequately addresses thermal fatigue and transients. In particular, the application should address (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 that could undergo thermal expansion mismatch caused by the mixing of coolant flows at different temperatures. The staff should verify that the application adequately addresses very high cycle fatigue due to thermal striping.

The staff should also consider potential thermal transients (including startup and shutdown) and the impacts of such transients on reactor components that are not addressed through ASME BPVC design rules. For example, operational experience has shown that thermal transients in HTRs can loosen shrink-fit components (NRC, 2019). The staff should confirm that the applicant

has considered 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, the applicant has addressed the synergistic effects of thermal fatigue, vibratory fatigue, and creep-fatigue whenever applicable.

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

This section identifies materials issues that are likely to apply to a variety of reactor designs, coolants, and materials. The discussion of these design-neutral issues reflects the current state of knowledge. As additional operating experience and laboratory testing data become available, the way these issues should be addressed may change, and new issues may be identified. In the meantime, the staff should evaluate whether applicants have adequately addressed these issues as appropriate for their specific application and design.

Metallic Materials

The staff should ensure that the metallic materials to be used for structural components have been tested under conditions representative of the anticipated operating environment in terms of temperature, impurity levels, and the potential for degradation mechanisms resulting from the reactor environment. The staff should review metallic cladding (ORNL, 2021) to verify that the cladding is qualified in a representative environment, with consideration given to adherence to its metallic substrate and galvanic coupling. The staff should confirm that the applicant has considered appropriate mitigation strategies, performance monitoring, and surveillance programs to ensure that metallic materials and coatings continue to satisfy the design criteria under all relevant environmental conditions.

Dissimilar-Metal Welds

III-5 provides stress rupture factors to account for the reduced creep strength of welds for the six 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 shorter creep lifetimes than welds using only one of the metals involved (e.g., welds of ferritic low-alloy steel or austenitic alloy alone) (EPRI, 2020a; INL, 2024). Examples of metallurgical phenomena that can contribute to

the reduced lifetime of DMWs include differing coefficients of thermal expansion for the weld constituents and compositional gradients in different alloys driven by high-temperature solid-state diffusion. The staff should evaluate whether the potentially lower lifetimes of DMWs, particularly between ferritic low-alloy steels and austenitic alloys, have been adequately addressed. The staff should verify that the applicant has considered appropriate mitigation strategies, performance monitoring, and surveillance programs to ensure that components containing 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 et al., 2007; ORNL, 2018). The properties of SiC/SiC depend on 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.

RG 1.87, Revision 3, endorses the 2023 edition of III-5, including a subpart addressing nonmetallic composites. The staff should review the compatibility of composites with the coolant environment based on the factors discussed above. The staff should confirm that the applicant has considered appropriate performance monitoring and surveillance programs to ensure that SSCs fabricated with these composites continue to satisfy the design criteria.

Ceramic Insulation

The staff should evaluate whether the application adequately addresses the environmental effects of ceramic insulation. The staff should confirm that the application considers the 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 operation, anticipated operational occurrences, and accident conditions (Guidez and Prèle, 2017).

Gaskets and Seals

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

Advanced Manufacturing Technologies

The staff should evaluate whether an application containing AMT components considers (1) the differences between the AMTs and traditional manufacturing methods, (2) the safety significance of the differences identified, (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 qualification and performance of each AMT. It is particularly important that an application fully address AMT material performance at high

temperatures. Limited studies have shown that AMT materials may exhibit lower long-term creep, fatigue, and creep-fatigue performance values than wrought materials (INL, 2020; INL, 2021). The staff should confirm that the applicant has considered appropriate mitigation strategies, performance monitoring, and surveillance programs to ensure that SSCs fabricated by AMTs continue to satisfy the design criteria.

Reactor-Specific Guidance, Part 1: Molten Salt Reactors

This section identifies 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 tristructural isotropic (TRISO) fuel particle, which could be in a graphite compact or pyrolytic graphite sphere. In solid-fuel MSRs, the molten salt coolant may contain relatively small quantities of fission products. MSRs can use a fast neutron or thermal neutron spectrum, and both types of MSR designs operate at near-ambient pressures. Molten salt is generally corrosive to traditional metallic SSCs. Corrosion can be increased 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 (ORNL) is the only reported example of an operational power MSR (EPRI, 2019a).

This section offers details on the design- and environment-specific aspects of the degradation mechanisms described in the section “General Degradation Mechanisms” above. The staff should evaluate whether applicants have adequately addressed these mechanisms and have presented appropriate 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, and inspection programs address any potential chemical compatibility issues, as applicable.

The carburization of structural alloys can increase the degradation rates of these materials (Chan et al., 2018; NRC, 2021d). The staff should review interactions between graphite and metals to confirm that qualification testing, monitoring, surveillance, and/or inspection programs adequately address potential carburization.

The staff should evaluate whether the application adequately addresses 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 addresses the potential for increased corrosion caused by the contact of graphite with metallic materials. Increased corrosion of 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 addresses whether the porosity or grain size of the graphite components allows for salt infiltration. If so, the application should

assess whether salt intrusion could cause any cracking or flaw generation in the graphite, thereby shortening the effective life of the graphite.

The staff should evaluate whether the application adequately addresses the potential for molten salt to accelerate wear, abrasion, and/or erosion between graphite components. The staff should verify that the applicant has 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 addresses the potential for additional degradation concerns in liquid-fuel MSR when the fissile material is dissolved in the coolant. Fission products will also contribute to the contaminants in the liquid salt, and the application should consider their effects on materials wetted by the salt.

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

Radiation damage to molten salt could increase its corrosivity due to radiolytic decomposition of the salt over certain temperature ranges, which may undermine structural performance. Recombination rates have been 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 addresses this issue.

The staff should evaluate whether the application adequately addresses whether corrosion products from structural alloys could affect degradation rates for SiC/SiC composites used as structural components (excluding fuel, which is beyond the scope of this guidance). For example, chromium (Cr) carbides may be formed by Cr^{3+} from Hastelloy N, which may cause accelerated corrosion of SiC (ORNL, 2018).

The staff should evaluate whether the applicant has adequately considered the realistic environment of the reactor when assessing the corrosion rates and behaviors of structural alloys. For example, many experimental demonstrations of corrosion rates for structural alloys are performed in static environments, while thermal convection loop tests can show different results (Pint et al., 2025). Flowing thermal convection loops of FLiNaK [eutectic mixture of lithium fluoride, sodium fluoride, and potassium fluoride] salts have shown mass gain on specimens located in the lower-temperature portion of the convection loop, along with much more severe mass loss in the hottest portions, compared to static tests. This could point to a saturation effect in static exposures. Mass loss in FLiBe [mixture of lithium fluoride and beryllium fluoride] salts is also generally greater in flowing than in static conditions. ORNL (2023) gives an overview of various molten salt corrosion testing methods. The staff and applicants should be aware of the limitations of each experimental approach to appropriately evaluate the applicability of any corrosion data presented in an application.

The staff should confirm that the applicant has 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 addresses the effects of salt composition on the degradation of metallic and nonmetallic materials due to molten salt, which may undermine structural performance by increasing the likelihood of crack initiation or by reducing strength or ductility. The staff should consider the effects of oxidizing impurities and of reducing agents. Oxidizing impurities include fission products (although these may be limited in a salt-cooled HTR design), water, air, and, for salts that contain lithium, tritium (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 methods could include the following:

- gas phase control (e.g., HF/H₂)
- major metal control (e.g., Be²⁺/Be)
- dissolved salt control (e.g., U⁴⁺/U³⁺ or Ce³⁺/Ce⁴⁺)

The staff should verify that the applicant has considered appropriate mitigation strategies, performance monitoring, and surveillance programs to ensure that salt composition does not exceed allowable limits, so 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 weight percent lead (Pb) and 55.5 weight percent bismuth). The sodium-cooled fast reactor (SFR) has been in use for decades at the experimental, prototype, and commercial scales. The lead fast reactor uses liquid lead or LBE as the coolant (EPRI, 2019b), and the design concepts span operating temperatures of 550–800°C. To date, operational experience with lead fast reactors is limited to the development of LBE-cooled reactors for the Alfa-class submarines operated by the Soviet Union from 1967 to 1983 (EPRI, 2019b; Alemberti et al., 2014; IRSN, 2012). In 2021, however, construction began on the first prototype lead-cooled reactor, the BREST-OD-300, in the Russian Federation (Proctor, 2021).

This section offers details on the design- and environment-specific aspects of the degradation mechanisms described in the section “General Degradation Mechanisms” above.

Sodium Coolant

Below are additional degradation considerations likely to apply to sodium-cooled liquid metal reactors. The staff should evaluate whether the applicant has adequately addressed these considerations and has considered appropriate mitigation strategies, performance monitoring,

and surveillance programs to ensure that SSCs in all environments continue to satisfy the design criteria.

Caustic Stress-Corrosion Cracking

The staff should evaluate whether the application adequately addresses 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 Phénix reactor demonstrated that 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, so as to minimize the potential for caustic SCC, and that the applicant has considered appropriate mitigation strategies, performance monitoring, and surveillance programs to minimize the potential for caustic SCC and ensure that component integrity satisfies the design criteria.

Exothermic Reactivity with Water

The staff should evaluate whether the application adequately addresses the potential for molten sodium to react with water or moisture in the air. 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 the applicant minimizes the potential for sodium-water reactions through design, and that the applicant has 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 addresses the temperature, flow rate, and impurity limits for the sodium coolant (notably, for oxygen and carbon), since these parameters significantly affect the corrosion rate of metallic components in contact with the sodium coolant (Thorley and Tyzack, 1967; ANL, 2017; NRC, 2021e), which may undermine structural performance by increasing the likelihood of crack initiation or reducing strength or ductility. Studies conducted with varying 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 the specific reactor design (ANL, 1978; Hanford, 1980; NRC, 2021e).

The staff should be aware that data from short-term (2,000-hour) static testing indicate that SiC/SiC may be resistant to corrosion from high-purity sodium (1 weight part per million) at 550°C, but the corrosion resistance decreases as the concentration of oxygen in the sodium increases (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., for flow rate and impurities, in particular oxygen).

Liquid Metal Embrittlement

The staff should evaluate whether the applicant has 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 strategies, performance monitoring, and surveillance programs to manage LME are adequate.

Carburization and Decarburization

The staff should be aware of the potential sources of sodium impurities, such as carbon in the sodium. Carburization and decarburization are both well-documented in sodium reactors; for example, carbon impurities contained 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 ensure that the potential for carburization and decarburization of structural alloys is controlled and that measures are in place to maintain appropriate liquid sodium purity during the plant design life.

Lead Coolant

A “lead-cooled” reactor may use lead (T_{melt} , 327.5°C) or LBE alloy (T_{melt} , 123.5°C) as the coolant. Metallic elements used in structural alloys, including iron, nickel, and chromium, are all soluble in lead and LBE, and their solubility in either coolant depends strongly on 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, at the same temperature, LBE is typically more corrosive than pure lead (NEA, 2015).

Below are additional degradation considerations likely to apply to lead-cooled liquid metal reactors. The staff should evaluate whether applicants have adequately addressed these considerations and have presented appropriate plans to monitor, evaluate, and mitigate degradation.

Corrosion at Higher Temperatures

Many alloys, including those approved for III-5 use, can experience high corrosion rates in a lead or LBE environment at higher temperatures, defined here as temperatures greater than 550°C. For example, rapid corrosion of Type 316 stainless steel occurs above 550°C even with tight oxygen control because of the transition from a protective to a nonprotective oxide (EPRI, 2019b). The staff should evaluate the 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 the flow velocity of lead or LBE coolant increases the corrosion rate of structural materials in contact with the coolant (Ballinger and Lim, 2003; Allen and Crawford, 2007; IRSN, 2012; Vogt and Serre, 2021). However, corrosion rates are not solely dependent on the coolant flow velocity; they also depend on temperature, material, and coolant impurities. Hence, higher

flow velocities may be acceptable, especially for lower operating temperatures. The staff should review temperature, flow velocities, and dissolved oxygen to accurately assess 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. It 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; NEA, 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 the applicant's material selection and supporting data to ensure that the potential effects of the lead or LBE environment on the mechanical properties of alloys 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 on Type 316L stainless steel in LBE coolant increases with increasing percentages of cold work (Klok et al., 2017).

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

Nonmetallic Materials

SiC/SiC has shown resistance to liquid metal corrosion up to 550°C in 2000-hour corrosion tests in flowing liquid LBE (Takahashi and Kondo, 2011). 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 means of ensuring satisfactory performance of structural materials in these reactors and is widely used in lead-based test facilities worldwide (Tarantino et al., 2021). Oxygen control refers to maintaining the oxygen concentration in the coolant within controlled limits. Corrosion rates at temperatures below 450°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 critical over the entire range of operating 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 and other detrimental effects (NEA, 2007; IRSN, 2012). The staff should evaluate whether the applicant has 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-Cooled Reactors

High-temperature gas-cooled reactors can use helium or carbon dioxide (CO₂) coolant; however, reactors that use CO₂ as the coolant, such as the Advanced Gas Reactor and Magnox reactors in the United Kingdom, are not currently expected to be deployed in the United States. Therefore, the following addresses only the 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 variants of the high-temperature gas-cooled reactor (HTGR), as described in EPRI (2020a). These designs have in common a reactor outlet temperature greater than or equal to 700°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).

This section offers details on the design- and environment-specific aspects of the degradation mechanisms described in the section “General Degradation Mechanisms” above. The staff should evaluate whether applicants have adequately addressed these mechanisms.

Creep-Rupture Strength

Service in a helium coolant environment has been shown to reduce the creep-rupture strength of structural alloys, in some cases to values below those specified in III-5 (Kim et al., 2013; Corwin et al., 2008; NRC, 2021c). The staff should ensure that the design analyses account for the potential for reduced creep-rupture strength within a helium coolant. The staff should confirm that the applicant has considered appropriate mitigation strategies, performance monitoring, and surveillance programs to ensure that SSCs that could be affected 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). Greater surface roughness generally increases emissivity but has a negative effect on fatigue performance. The staff should closely review designs that increase surface roughness to increase emissivity to ensure that impacts on fatigue performance are appropriately addressed and the required emissivity values will be achieved.

Thermally grown surface oxide or carbide can affect emissivity on both the inside and outside of HTGR reactor pressure vessels. Within the reactor pressure vessel and primary-loop SSCs, the 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, oxidizing impurities, and abrasion or coating of metallic surfaces by graphite dust to affect the emissivity of structural alloys (NRC, 2021c). The staff should confirm that the applicant has considered appropriate mitigation strategies,

performance monitoring, and surveillance programs to ensure that SSCs affected by these factors 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 depends on the graphite grade, temperature, and coolant impurities; in particular, coolant impurities may decrease the coefficient of friction (NRC, 2021g).

Graphite Dust and Debris

The staff should verify that the applicant has 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 and can hinder complete movement of the fuel or the control rod; also, graphite dust can agglomerate on piping, clogging the flow of helium (NRC, 2002). Operational experience has demonstrated that graphite dust can 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). In addition, graphite dust can carry absorbed fission products if fuel failure has occurred. Some previous pebble bed designs have experienced significant debris generation from broken fuel elements (pebbles) (NRC, 2025d). The staff should confirm that the applicant has considered appropriate mitigation strategies, performance monitoring, and surveillance programs to ensure that graphite dust and debris are 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 report shows typical concentrations of these impurities in previously operating very-high-temperature reactors (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 . Corrosion reactions with alloys may also produce CO and H_2 . CO_2 degassing from graphite converts to 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 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. HTGRs operated to date have maintained the total impurity levels in the helium below 10 parts per million to minimize these effects (INL, 2006). The staff should evaluate the coolant gas composition to confirm that the potential for carburization, decarburization, and oxidation of structural alloys is controlled and that appropriate measures are in place to maintain appropriate gas composition during the plant design life.

Silicon Carbide and Silicon Carbide Composites

The staff should evaluate the use of SiC and SiC/SiC composites in HTGRs and consider the potential 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 et al., 2021; EPRI, 2020a).

Lubricants

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

IMPLEMENTATION

The staff will use the information in this ISG to review non-LWR applications for construction permits and operating licenses under 10 CFR Part 50, combined licenses, standard design approvals, design certifications, and manufacturing licenses under 10 CFR Part 52, and construction permits, operating licenses, combined licenses, standard design approvals, design certifications, and manufacturing licenses under 10 CFR Part 53 that propose to use materials allowed under III-5 or other commercial codes and standards for the design and construction of mechanical components.

BACKFITTING, FORWARD FITTING, AND ISSUE FINALITY

The NRC staff may use this ISG 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

Discussion to be provided in the final ISG.

PAPERWORK REDUCTION ACT

This ISG provides voluntary guidance for implementing the mandatory information collections in 10 CFR Parts 50, 52, and 53 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, 3150-0151, and 3150-0274. Send comments regarding this information collection to the FOIA, Library, and Information Collections Branch at Infocollects.Resource@nrc.gov and to the OMB reviewer at: OMB Office of Information and Regulatory Affairs (3150-0011, 3150-0151, and 3150-0274), Attn: Desk Officer for the Nuclear Regulatory Commission, 725 17th Street, NW, Washington, DC, 20503.

PUBLIC PROTECTION NOTIFICATION

The NRC may not conduct or sponsor, and a person is not required to respond to, a collection of information unless the document requesting or requiring the collection displays a currently valid OMB control number.

FINAL RESOLUTION

The staff will transition the information and guidance in this ISG into RG 1.87 or the 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 Content of Applications for Materials Performance for Advanced Non-Light-Water Reactors
- B References

APPENDIX A

Content of Applications for Materials Performance for Advanced Non-Light-Water Reactors

1. Introduction and Purpose

The purpose of this appendix is to provide guidance on information the U.S. Nuclear Regulatory Commission (NRC) staff generally has found necessary to perform its review of materials of construction and the performance of passive structures, systems, and components (SSCs) for non-light-water reactors (non-LWRs) under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, “Domestic Licensing of Production and Utilization Facilities,” and 10 CFR Part 52, “Licenses, Certifications, and Approvals for Nuclear Power Plants.” As noted above, the information necessary for reviewing the application will depend on many factors. This guidance is intended to clarify what has previously been found acceptable up front for the content of applications in this area, enabling more efficient staff review as well as regulatory consistency and predictability. The appendix incorporates lessons learned by the NRC staff from recent non-LWR licensing reviews. It is also informed by preapplication engagement, topical reports, and information that has historically been supplied for light-water reactor (LWR) applications. The staff encourages applicants to be aware of the information in this appendix. This will facilitate the exchange of information between applicants and the NRC staff, assure high-quality licensing applications, and increase the efficiency of licensing reviews.

2. Regulatory Requirements and Safety Classification

The requirements for the content of applications are established in 10 CFR Part 50, 10 CFR Part 52, and 10 CFR Part 53, “Risk-Informed, Technology Inclusive Regulatory Framework for Commercial Nuclear Plants.”

Parts 50 and 52

The regulation at 10 CFR 50.34(a) requires that a construction permit (CP) application include a preliminary safety analysis report (PSAR) and describes the required contents of that report, while 10 CFR 50.34(b) requires that an operating license (OL) application include a final safety analysis report (FSAR) and describes the required contents of that report. The regulations in 10 CFR Part 52 provide for a one-step licensing process based on a combined license (COL) application. The requirements for the technical content of such applications can be found in 10 CFR 52.79 and 10 CFR 52.80. Similar final design technical content of application requirements exist in Part 52 for standard design certification (DC) and standard design approval (SDA) applications in 10 CFR 52.47 and 52.137, respectively. For manufacturing license applications, the requirements for the technical content of such applications can be found in 10 CFR 52.157 and 52.158.

Part 53

10 CFR Part 53 contains similar requirements on the technical content of applications. For CP applications these can be found in 10 CFR 53.1309 and 53.1312. For OL applications these can be found in 10 CFR 53.1369 and 53.1372. For COL applications these can be found in 10 CFR 53.1416 and 53.1419. For SDA applications these can be found in 10 CFR 53.1209 and 53.1210. For DC applications, these can be found in 10 CFR 53.1239 and 53.1241. For ML applications these can be found in 10 CFR 53.1279 and 53.1280.

Further details from these regulatory requirements related to preliminary and final design information are discussed below.

License Type

Construction Permit Application

In relation to materials performance, 10 CFR 50.34(a)(3) requires a CP application to include a PSAR that includes the principal design criteria (PDC) for the facility, the design bases and how they relate to the PDC, and “[i]nformation relative to materials of construction, general arrangement, and approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety.” In addition, 10 CFR 50.34(a)(4) requires the PSAR to include a preliminary analysis and evaluation of the design and performance of SSCs of the facility with the objective of assessing the risk to public health and safety resulting from operation including determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and the adequacy of SSCs provided for the prevention of accidents and the mitigation of the consequences of accidents.

In 10 CFR Part 53, the requirements for the technical content of CP applications appear at 10 CFR 53.1309, “Contents of applications for construction permits; technical information.” These require that a CP application include a PSAR. The PSAR for a commercial nuclear plant must include design information equivalent to that required by 10 CFR 53.1239(a)(2)-(a)(21), (a)(23), and (a)(26)-(a)(27), “Contents of applications for standard design certifications; technical information,” for a standard design certification except as specified in 10 CFR 53.1309(a)(2). This includes information on:

- codes and standards (C&S) (equivalent to the information required by 10 CFR 53.1239(a)(11) which cites the requirements of 10 CFR 53.440(b)),
- materials (equivalent to the information required by 10 CFR 53.1239(a)(12) which cites the requirements of 10 CFR 53.440(c)), and
- an integrity assessment program (equivalent to the information required by 10 CFR 53.1239(a)(13) which cites the requirements of 10 CFR 53.440(d)).

Finally, 10 CFR 53.1312(b) modifies the information required to be in the PSAR for CP applications that cite an early site permit, standard design approval, standard design certification, or ML issued under Part 53. Table 1 in this appendix gives more details on a permissible way to provide this information for SSCs of different safety classifications.

Operating License/Combined Operating License/Standard Design Approval/Standard Design Certification/and Manufacturing License Application

In relation to materials performance for an OL application, 10 CFR 50.34(b) requires OL applicants to submit an FSAR which must include “[a] description and analysis of the [SSCs] of the facility, with emphasis upon performance requirements, the bases...upon which such requirements have been established, and the evaluations required to show that safety functions will be accomplished.” It also requires “[a] final analysis and evaluation of the design and performance of [SSCs] with the objective stated in [50.34(a)(4)] and taking into account any

pertinent information developed since the submittal of the preliminary safety analysis report,” as well as “[plans] for conduct of normal operations, including maintenance, surveillance, and periodic testing of [SSCs].” For COL applications, 10 CFR 52.79, “Contents of applications; technical information in final safety analysis report,” requires that COL applications include an FSAR which is required to contain information similar to that required for CP and OL applications under 10 CFR 50.34(a)–(b). Similar content of application requirements appear in Part 52 for SDA, DC, and ML applications in 10 CFR 52.137, 10 CFR 52.47, and 10 CFR 52.157, respectively.

In 10 CFR Part 53, the requirements for the technical content of OL applications appear at 10 CFR 53.1369, “Contents of applications for operating licenses; technical information.” These require that an OL application contain an FSAR. The FSAR for a commercial nuclear plant must include final design information equivalent to that required 10 CFR 53.1239(a)(2)–(a)(7), (a)(9), (a)(11)–(a)(12), (a)(14)–(a)(21), (a)(23), and (a)(25) for a standard design certification except as specified by 10 CFR 53.1369(b)(2). This includes information codes and standards (equivalent to the information required by 10 CFR 53.1239(a)(11) which cites the requirements of 10 CFR 53.440(b)) and materials (equivalent to the information required by 10 CFR 53.1239(a)(12) which cites the requirements of 10 CFR 53.440(c)). Further, under 10 CFR 53.1369(d) the FSAR must include a description of an integrity assessment program that addresses the elements described in 10 CFR 53.870.

The corresponding requirements for COLs appear in 10 CFR 53.1416, “Contents of applications for combined licenses; technical information.” These state that a COL application must contain an FSAR which must contain information “at a level of detail sufficient to enable the Commission to reach a final conclusion on all safety matters that must be resolved by the Commission before issuance of a COL.” As relevant here, the FSAR for a commercial nuclear plant must include design information covering similar topics to that discussed for an OL application. Similar content of application requirements appear in Part 53 for SDA, DC, and ML applications in 10 CFR 53.1209, 10 CFR 53.1239, and 10 CFR 53.1279, respectively. One relevant difference for SDA, DC, and ML compared to OL and COL is that these application types cross-reference or include 10 CFR 53.1239(a)(13), which cites the requirements of 10 CFR 53.440(d), rather than requiring a description of an integrity assessment program that addresses the elements described in 10 CFR 53.870. Table 2 of this appendix gives more details on a permissible way to provide this information for SSCs of different safety classifications.

Safety Classification

Safety classification is the process by which the SSCs within a design are categorized based on their importance to safety.

Parts 50 and 52

The Licensing Modernization Project (LMP) methodology described in NEI 18-04, Revision 1, and endorsed by Regulatory Guide (RG) 1.233, “Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors,” defines three safety classification categories for SSCs: safety-related (SR), non-safety-related with special treatment (NSRST), and non-safety-related with no special treatment (NST) (NEI 2019; NRC, 2020). This appendix will use the terms SR, NSRST, and NST as defined in the LMP methodology endorsed in RG 1.233 when discussing risk-informed safety classifications for applications submitted under 10 CFR parts 50 and 52.

In the traditional deterministic process used for operating power LWRs and non-LWRs that have recently been licensed by the NRC as Class 104 research and test reactors, SSCs are determined to be as SR or non-safety-related (NSR) based on the definition of SR SSCs found in 10 CFR 50.2, “Definitions,” or an alternative definition.¹ In addition, the regulations in 10 CFR Part 50 use the term “important to safety” to refer to a broader group of SSCs (encompassing all SR SSCs and some NSR SSCs) that are needed to meet requirements such as the PDC, fire protection requirements, environmental qualification requirements, and maintenance requirements. This phrase goes beyond the definition of SR SSCs in 10 CFR 50.2.² *For SSCs that are classified as NSR under the traditional deterministic process but that perform important-to-safety or risk-significant functions, applicants should follow the guidance for NSRST and safety-significant SSCs in this appendix. For NSR SSCs that do not perform any important-to-safety or risk-significant functions, applicants should follow the guidance for NST SSCs in this appendix.*

Part 53

Similarly to the LMP methodology endorsed in RG 1.233, 10 CFR Part 53 defines three safety classification categories: SR, non-safety-related but safety significant (NSRSS), and non-safety-significant. These categories are defined in 10 CFR 53.020.

Requirements under 10 CFR Part 53 related to safety classification of SSCs were developed based on the same principles and concepts from SSC classification process described in NEI 18-04 and endorsed in RG 1.233. Because 10 CFR Part 53 and the process endorsed in RG 1.233 use similar constructs for the safety classification of SSCs, consideration of issues related to materials performance and compatibility are likewise similar. This appendix uses the terms SR, NSRSS, and non-safety-significant as defined in 10 CFR Part 53 when discussing Part 53 safety classification of SSCs. Accordingly, Part 53 applicants should note that *the guidance for NSRST and safety-significant SSCs in this appendix is applicable to NSRSS SSCs under Part 53. In addition, the guidance for NST SSCs in this appendix is applicable to non-safety-significant SSCs under Part 53.*

Safety-Related Structures, Systems, and Components

SR SSCs are defined in 10 CFR 50.2 as well as under the LMP methodology endorsed in RG 1.233. The definition from 10 CFR 50.2 is applied in regulatory requirements throughout 10 CFR Part 50. Under the LMP methodology endorsed in RG 1.233, SR SSCs are identified through a risk-informed process, further details of which can be found in NEI 18-04 and RG 1.233. In this appendix, “SR SSCs” refers to items classified as SR under either the LMP

¹ Guidance in NUREG-1537, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors,” issued February 1996, has been used for research and test reactors (including recently issued non-LWR CPs) and does not provide specific guidance on safety classification of SSCs. General practice for nonpower reactors and non-LWRs has been to use an alternative definition for SR SSCs to that in 10 CFR 50.2, because the latter definition is specific to water-cooled power reactors.

² The *Federal Register* notice “Determining Which Structures, Systems, Components and Functions Are Important to Safety” ([87 FR 11986](#); March 3, 2022) provides greater detail on the history of the term “important to safety.” The regulations in 10 CFR 50.69, “Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors,” also use the term “nonsafety-related SSCs” to define categories of SSCs based on risk (categories RISC-2 and RISC-4).

methodology endorsed in RG 1.233, the requirements of 10 CFR Part 53, or a deterministic process under 10 CFR Parts 50 or 52.

Because SR SSCs have the highest safety classification and perform required safety functions, the application should provide the greatest level of detail and the strongest basis for the design, construction, and in-service performance of SR SSCs. Tables 1 and 2 below describe the information for SR SSCs that is acceptable at the preliminary and final design stages.

Non-Safety-Related with Special Treatment and Non-Safety-Related but Safety-Significant Structures, Systems, and Components

NSRST SSCs are an intermediate category of SSCs in the LMP methodology endorsed in RG 1.233. NEI 18-04 and RG 1.233 give further details on how NSRST SSCs are identified under the LMP methodology. The term “special treatment” for NSRST SSCs is used consistent with RG 1.201, “Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance,” to refer to “those requirements that provide increased assurance beyond normal industrial practices that [SSCs] perform their design-basis functions.” Normal industrial practices are generally understood to be consistent with industrial or nonnuclear codes and standards. Therefore, special treatments are enhancements beyond the requirements of typical industrial codes and standards for a given SSC.³

Consistent with the definition in the LMP methodology endorsed in RG 1.233, NSRST SSCs are non-safety related SSCs relied on to perform risk-significant functions or relied on to perform functions requiring special treatment for defense in depth adequacy. Applicants should provide reasonable confidence of these SSCs’ performance. In general, the application should provide an appropriate level of detail and basis for the design, construction, and in-service performance of NSRST SSCs. Tables 1 and 2 in this appendix describe the information for NSRST SSCs that is acceptable at the preliminary and final design stages. Similar information is acceptable for NSR SSCs that are important to safety or risk significant under the traditional deterministic classification process, as described above. NSRSS SSCs under Part 53 are those SSCs that are not SR but are relied on to achieve adequate defense in depth or perform risk-significant functions. NSRSS SSCs are required to meet the provisions under 10 CFR 53.460 with respect to special treatments.

Non-Safety-Related with No Special Treatment and Non-Safety-Significant Structures, Systems, and Components

Under the LMP methodology endorsed in RG 1.233, any SSCs that are not identified as SR or NSRST are NST and are considered not to be safety significant and are not subject to special treatments. The regulations generally do not require significant information on NST SSCs. Usually, the staff’s review of these SSCs is focused on their potential to interfere with the ability of safety-significant and important-to-safety SSCs to perform their credited safety functions. Tables 1 and 2 below describe the information NST SSCs that is acceptable at the preliminary and final design stages. Similarly, the regulations do not requirement significant information

³ The LMP methodology endorsed in RG 1.233 also indicates that NSRST SSCs should be subject to a standard of reasonable confidence that they can perform their functions. The term “reasonable confidence” was introduced in the rulemaking for 10 CFR 50.69, for which the statements of consideration (69 FR 68047; November 22, 2004) included the following: “Conversely, for SSCs that do not significantly contribute to plant safety on an individual basis, this approach should allow an acceptable, though reduced, level of confidence (i.e., ‘reasonable confidence’) that these SSCs will satisfy functional requirements.”

about NSR SSCs that are not important to safety or risk significant under the traditional deterministic classification process. The regulations in 10 CFR Part 53 also do not require significant information about non-safety-significant SSCs. This type of SSC can be addressed under normal licensee programs for commercial grade equipment and typical industry practices for general plant design and maintenance.

3. Key Application Content Areas

This section describes key information that the staff should review to determine whether the application meets the regulatory requirements in 10 CFR Parts 50, 52, or 53 for the content of applications related to passive SSC materials performance. Section 2 of this appendix describes the requirements in 10 CFR Parts 50, 52, and 53 for the content of applications for CPs, OLs, COLs, SDAs, DCs, and MLs. These requirements can be understood as specifying two categories of information, the first describing the SSC design and operating environment, and the second describing the analyses or methodologies to evaluate and monitor passive SSC materials performance to provide assurance that the SSCs will perform as designed to assure reactor safety. Section 4 of this appendix discusses the licensing stage at which each type of information should be available for staff review.

Description of Design and Operating Environment

To meet the content of application requirements in 10 CFR Parts 50, 52, or 53 and to provide sufficient detail to evaluate their performance, the application should address the following areas.

Identification of Materials of Construction

The identification of materials is critical to providing assurance that safety-significant SSCs are compatible with their planned operating environment and can reliably perform their safety functions throughout their service life. Material properties and performance attributes, such as susceptibility to high-temperature creep and stress relaxation cracking, are specific to the alloy and grade of the material. Therefore, the precise identification of materials of construction for safety-significant SSCs is the starting point for all content of applications related to materials of construction and passive SSC performance.

Design information on materials of construction should generally include the materials, the fabrication methods used to produce components for service, and the SSC geometry. For metallic components, the materials of construction should include base materials, joining materials (such as welds or bolting), and coatings or other surface treatments that may be applied. Identification of materials should include chemistry (or material specification), the product form, joining methods such as welding (weld filler metal and welding process) or bolting, cold working, and other post-fabrication processing.

For graphite structural components, identification of materials should include the graphite grade, material specification, and a methodology for generating necessary properties for design. The latter is important because the material properties of various grades of graphite, unlike those of metals used in construction, are not already established in codes and standards. The application should provide sufficient information on the properties of graphite components to characterize their material behavior well enough to confirm that the components will perform their safety functions, with the appropriate confidence level. Qualification should account for material variance, including behavior within a billet, between billets, and between production

lots. Generally, graphite qualification should yield information on properties sufficient to parameterize a viscoelastic material model that includes irradiation creep considerations. The material properties of graphite materials can significantly affect an SSC's performance and susceptibility to degradation; therefore, the application should address them appropriately, consistent with the licensing stage and safety significance of the SSC.

Identification of Environmental Conditions

Environmental conditions are also vital to demonstrating that passive SSCs will reliably perform their safety functions. Environmental conditions may include temperature, chemistry, radiation, and stress (e.g., system pressure, residual stress, thermal stress), which should be identified for normal operating conditions as well as off-normal or accident conditions. Many advanced reactor applicants are focused on controlling coolant chemistry, or rely on cleaning or lubrication, to prevent material degradation. The details of these procedures should be adequately described commensurate with their importance for ensuring materials performance.

Information on normal operating conditions is important to understanding the aging effects, such as creep, fatigue, corrosion, or embrittlement, that may occur during exposure to the operating environment for the licensed period of operation. For off-normal or accident conditions, the application should contain a technical basis supported by testing or analysis to demonstrate that each safety-significant SSC can reliably perform its intended function (with degraded properties due to long-term aging in normal conditions) during the event but need not contain an analysis of performance under long-term exposure to these abnormal conditions.

Identification of Codes and Standards

Applicants should identify any C&S used for the design and construction of safety-significant SSCs. C&S may also be used to establish the attributes of SSC inspection and monitoring programs to demonstrate conformance with other design criteria, such as (for Part 50 and 52 applications) ARDC 32 and SFR-DC 77, under which the reactor should be designed to permit inspection of pressure boundary components when this would be necessary to ensure their structural integrity. While not required by regulation for non-LWRs, the use of NRC-endorsed nuclear C&S may support review efficiency. RG 1.87, "Acceptability of ASME Code, Section III, Division 5, 'High Temperature Reactors,'" and 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," endorse American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC), Section III, Division 5 (III-5), and Section XI, Division 2 (XI-2), respectively, which are related to the design, construction, and in-service inspection and monitoring of passive SSCs for high-temperature reactors. In addition, Section 1.1.5 of DANU-ISG-2022-01, "Review of Risk-Informed, Technology-Inclusive Advanced Reactor Applications—Roadmap," issued March 2024, provides guidance for applications under 10 CFR Parts 50 and 52 on identifying consensus C&S in a licensing application and on justifying departure from the identified C&S or the RGs endorsing them if they are endorsed (NRC, 2024b). Although DANU ISG-2022-01 is not applicable to 10 CFR Part 53, it may provide useful insights for applications under Part 53.

In addition to nuclear C&S endorsed by RG, there are many non-nuclear C&S, which have not been endorsed by the NRC, that applicants may propose to use for the design and construction of safety-significant SSCs. These include, but are not limited to, ASME BPVC Section VIII,

“Rules for Construction of Pressure Vessels,” Division 1, “Rules for Construction of Pressure Vessels,” or Division 2, “Alternative Rules for Construction of Pressure Vessels,” for vessels, and ASME B31.1, “Power Piping,” or B31.3, “Process Piping,” for piping (ASME, 2020a; ASME, 2020b). RG 1.87 provides guidance on how to identify and justify the use of C&S for safety-significant SSCs, while this appendix focuses on the type of information and level of detail that an application should include to sufficiently describe the use of C&S. If an applicant chooses to use Section VIII for safety-significant SSCs, the staff should review the relevant technical letter reports on the potential implementation of Section VIII (NRC, 2025b; NRC, 2025c). In many cases, if the applicant is using endorsed C&S in the type of application for which it has been endorsed, then the application will require less detail, because the technical evaluation of the C&S has already been performed and documented in an NRC RG. For non-endorsed C&S, more information is needed to enable the staff to evaluate its appropriateness for a specific use case.

Performance Evaluation and Monitoring

The SSC design and operating environment define the potential for materials degradation during the reactor operating life, as described in the main body of this interim staff guidance (ISG). Any reactor technology (LWR or non-LWR) creates materials performance challenges due to exposure to various environmental conditions (e.g., temperature, chemistry, irradiation). Non-LWRs present different challenges from LWRs because of their higher operating temperatures and use of different types of coolants (e.g., molten salt, sodium, inert gas). Some non-LWRs operate in temperature ranges with significant creep effects, which creates a time dependency in the allowable stresses used for design, because creep strength is reduced by longer times at elevated temperatures. In addition, non-LWR coolant environments may increase susceptibility to material corrosion and other chemically driven degradation mechanisms. To meet the content of application requirements in 10 CFR Parts 50, 52, and 53 and ensure adequate materials performance, the application should address the areas described below.

To ensure passive materials performance, applicants should appropriately identify potential degradation mechanisms based on the materials of construction and environmental conditions for safety-significant SSCs. These mechanisms should then be assessed by the applicant to ensure that they are adequately managed. The main body of this ISG provides guidance on the review of environmental compatibility in areas not covered by design and construction C&S. The environmental compatibility assessment should ensure that adequate consideration is given to several key topics related to materials degradation and performance:

- combined effects of multiple degradation mechanisms or stressors (e.g. temperature, chemistry, irradiation) on particular SSCs,
- inherent uncertainty from long-term combined effects that cannot be captured in short-term, separate effects testing,
- less than ideal conditions (e.g. local chemistry variations, chemistry transients, residual stresses, materials variability), and
- the potential for unexpected degradation due to lack of knowledge or experience, particularly over extended operating periods.

Licensing applications, whether for a preliminary design (CP) or a final design (OL, COL, ML, SDA or DC), should address environmental compatibility and materials performance for safety-significant SSCs to provide assurance of adequate passive SSC performance, considering the safety function, material, and operating environment of the SSC, as well as all potential degradation mechanisms. Applicants should provide the level of information commensurate with the safety classification. NSRST SSCs are subject to the lower standard of reasonable confidence of performance of safety functions, consistent with their safety significance. Acceptable passive SSC materials performance is typically demonstrated using a combination of the following types of information.

Existing or Historical Experimental Data

Many non-LWR designs leverage existing or historical data from research programs conducted by U.S. Department of Energy national laboratories, universities, or international organizations. An applicant using such data should clearly explain how the data apply to its design and relate to SSC performance—for example, by demonstrating what would be allowable operating conditions. The applicant should justify that existing or historical data are of appropriate quality for their use. For example, ASME NQA-1 Subpart 4.2.3, “Guidance on Qualification of Existing Data,” provides several approaches to qualify existing data to the appropriate quality standard.

New Experimental Data Generated by Ongoing or Planned Testing

Many applicants have testing programs designed to address certain key materials performance issues, such as chemical effects tied to the coolant or irradiation effects tied to design-specific irradiation conditions (e.g., fluence, temperature). If using such testing programs, the applicant should clearly explain how the data apply to its design (including consideration of rate effects due to accelerated testing) and how they will be used to provide assurance of materials performance. Quality assurance (QA) should be appropriately considered in the collection and use of experimental data to support the design basis. Regulatory Issue Summary (RIS) 2024-02, “Required Assessment of U.S. Department of Energy Laboratories by Licensees, Applicants, and Suppliers to Verify the Effective Implementation of Their Quality Assurance Programs,” provides the NRC’s position on the responsibility of licensees, applicants, and vendors for verifying the effective implementation of a national laboratory’s quality controls (NRC, 2024c). Although RIS 2024-02 is not applicable to 10 CFR Part 53, it may provide useful insights for applications under Part 53.

Operating Experience

Some materials performance topics may be supported by operating experience from prior test or commercial reactors. The extent to which operating experience can be relied upon depends on the degree of similarity between the design and operating conditions of the new reactor and those of the reactor or system to which it is being compared, as well as the intended use of the data or observations. The application should include a justification based on this comparison, with a level of detail consistent with the licensing stage and the safety significance of the SSC. This justification may include the material, the SSC’s geometry and function, the operating time and environment, and the type of data or information collected to support the conclusions from the operating experience.

For example, if a section of piping in a test reactor operated for 10 years without leaking, that experience would indicate that significant degradation to create a through-wall leak had not occurred in that time period. This basis would be strengthened if there were post-service

inspections or destructive testing confirming that no wall thinning, cracking, or change in material properties had occurred, to help demonstrate a lack of active degradation over that period. However, the 10 years of experience without leakage alone would not demonstrate that no active degradation had occurred, only that no degradation had progressed to the point of a through-wall defect causing leakage.

Analysis

Various analytical and modeling tools can help demonstrate acceptable materials performance. These range from comparatively simple cycle counting analyses to assess fatigue in accordance with C&S rules to highly complex finite element modeling to assess stress corrosion cracking susceptibility. Analytical approaches should be validated by appropriate experience and experimental data, commensurate with how they are being applied. Various NRC documents provide guidance on acceptable methods for doing this. For example, section 2.4 of RG 1.245, "Preparing Probabilistic Fracture Mechanics Submittals," issued January 2022 (NRC, 2022a), provides guidance on software QA and verification and validation for probabilistic fracture mechanics submittals. In addition, RG 1.203, "Transient and Accident Analysis Methods," issued December 2005 (NRC, 2005), and RG 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance," issued May 1989 (NRC, 1989), provide guidance on verification, validation, and uncertainty quantification (including best estimate plus uncertainty) for systems-level codes used for NRC licensing. These documents are only applicable to some regulations covered by this ISG but may provide useful insights for applications under regulations to which they are not applicable.

Performance Monitoring

The use of appropriate design and construction C&S should provide a strong base of confidence, but known and unknown materials degradation mechanisms can lead to SSC failure, particularly for long-term environmental effects that are difficult to address through experimental data or prior operating experience. Performance monitoring refers to identifying the occurrence or extent of degradation in SSCs through various means, such as nondestructive inspection (e.g., visual or ultrasonic examination), use of surveillance coupons to monitor changes in material properties (e.g., embrittlement or loss of strength), or sensors to detect off-normal conditions, such as leakage or unexpected vibration. Performance monitoring to confirm passive SSC performance should be a key element of managing uncertainty and demonstrating adequate performance during operation. An adequate performance monitoring program should ensure environmental compatibility and support design criteria such as ensuring that the reactor be designed to allow for inspection or monitoring of components for which this is necessary to ensure their integrity.

The description of a performance monitoring program should include information about how components or locations to be monitored will be selected, how it will be determined that the monitoring methods and frequencies are appropriate to detect potential degradation, and how criteria that establish acceptable or unacceptable conditions will be defined. Generally, the application of appropriate C&S for performance monitoring should provide all or most of the basis for the staff to evaluate the adequacy of the program description. XI-2 provides one method for developing a comprehensive aging management strategy, subject to NRC acceptance of the proposed program, as endorsed in RG 1.246. Under this method, an applicant develops strategies for inspecting, monitoring, and repairing SSCs throughout the design lifetime.

The staff should evaluate whether the applicant's performance monitoring program for ensuring component integrity and materials performance provides the following:

- direct evidence of the presence and/or extent of degradation
- a timely method of detecting novel or unexpected degradation
- confirmation of the continued adequacy of each SSC to fulfill its design and functional requirements over its lifetime

Direct evidence should be developed consistent with each expected degradation mechanism. In this context, direct evidence refers to an observation or measurement taken from the SSC itself, rather than an observation or measurement of a general system parameter (e.g., temperature or flow rate) from which the condition of the SSC is inferred. For example, a creep-fatigue monitoring system provides direct evidence of performance if it measures the material response to the creep-fatigue conditions (e.g., increases in strain or development of cracking). It does not provide direct evidence if it simply measures environmental conditions such as temperature and flow, then uses a model with inherent uncertainty to infer creep-fatigue performance.

For timeliness, the periodicity and sampling intensity for performance monitoring (e.g., the number of locations and/or SSCs examined) should be sufficient to provide a reasonable expectation that degradation will be detected before any loss of function. Finally, the program should provide confirmation of the continued adequacy of the SSC to perform necessary functions, including confirmation of any assumptions about SSC performance used in risk-informed analyses. For example, performance monitoring may help confirm estimates of the probabilities of SSC failure for SSCs with limited operating experience.

Ultimately, the licensing application should provide sufficient information on the management of potential degradation mechanisms for each safety-significant SSC to demonstrate acceptable materials performance based on the safety function and safety classification of the SSC. Reviewers should consider all relevant information, including safety margins, defense in depth, consequences of loss of function, uncertainty, and other means of identifying degradation, to reach holistic conclusions about the appropriate assurance of passive SSC materials performance.

4. Information Expected for Licensing Applications

Docketed Information and Audits

Information needed to support regulatory findings must be docketed. This is typically done in the PSAR or FSAR but may also be through referenced topical reports or other docketed submittals, such as responses to requests for additional information or supplements to the application.

The NRC staff can perform regulatory audits to efficiently gain understanding, verify information, and/or identify information requiring docketing to support a licensing decision. Office Instruction LIC-111, Revision 2, "Regulatory Audits," dated December 19, 2024 (ML24309A281), provides guidance to NRC staff members who participate in regulatory audits of information developed, maintained, and submitted by licensees, preapplicants, applicants, and vendors. Audits allow the NRC staff to do the following:

- Better understand the detailed calculations, analyses, and bases underlying the formal application to verify docketed information.
- Identify additional information necessary to reach a licensing or regulatory decision that should be provided on the docket.
- Understand an area where the staff has identified potential concerns, so that the staff can issue clear requests for information to be added on the docket.

Tables 1 and 2 in this appendix discuss the types of information that should be available for audit of a CP applicant.

Guidance on Application Content Information on Passive SSC Materials Performance

Based on recent licensing experience, table 1 identifies the information generally used to make a safety finding at the preliminary design stage in a CP application, consistent with the requirements of 10 CFR 50.34(a) and 10 CFR 53.1309. Inclusion of this information in a CP application facilitates an efficient and timely licensing review. *Preapplication engagement and interactions with the NRC staff are not required* before the submittal of a licensing application, but are highly encouraged, particularly to identify and discuss novel, complex technical issues. This is discussed further in Appendix A to DANU-ISG-2022-01. Although DANU ISG-2022-01 is not applicable to 10 CFR Part 53, it may provide useful insights for applications under Part 53. Topics in table 1 recommended for preapplication engagement include the selection of C&S and materials qualification. For preapplication topical reports for which a safety evaluation would be incorporated by reference into the safety evaluation for a CP, the information should be equivalent to that for a CP.

Materials Qualification Plans

From past experience, one particularly useful form of preapplication engagement related to materials performance is the development of materials qualification plans for novel materials and methods, for supplemental environmental testing, or to demonstrate ASME code consistency. Materials qualification plans for those purposes generally have the following attributes:

- Novel materials and methods: The plan demonstrates how novel materials and manufacturing methods develop appropriate design allowables, as well as how construction rules (i.e., material, design, fabrication, examination, testing, overpressure protection, and QA requirements) are being met. The draft generic guidelines for advanced manufacturing technologies (NRC, 2021a), which address key topics such as QA, process qualification, and production process control and verification, may provide useful information on how to demonstrate materials performance for novel materials and manufacturing methods. That said, as noted in these documents, these draft guidelines do not represent official agency positions. The NRC staff has not determined that consistency with them is an acceptable way of meeting the regulatory requirements. Further, these draft guidelines were developed with for applications under 10 CFR Part 50, but the concepts discussed may be relevant to applications under 10 CFR Parts 52 and 53 as well.

- Environmental compatibility assessment and testing: The plan describes how appropriate data on environmental effects (e.g., temperature, coolant chemistry, irradiation) are available or will be generated to establish end-of-life properties for design. Preapplication engagement is encouraged to discuss how these data (whether already existing or to be obtained from planned testing) apply to the proposed plant and support reactor design and operating envelopes, including operating time and temperature, fluence, coolant impurity levels, and off-normal or accident conditions, as appropriate. The main body of this ISG provides guidance to the NRC staff on the review of environmental compatibility assessments for non-LWR designs.
- ASME code consistency: The plan demonstrates how data are collected in accordance with the code of construction, including any endorsement conditions if applicable. This is particularly relevant for data on nonmetallic materials, such as graphite and composite materials, for which III-5 generally calls for specific testing.

Table 1 Application Content Information on Passive SSC Materials Performance that Should be Available During CP Application Reviews of Preliminary Design

Key Area	Safety-Significant (SR or NSRST) SSCs	NST SSCs
Identification of materials of construction	The CP application should identify materials of construction for safety-significant SSCs, including SSC geometry, specific alloy or grade of material, weld metals, and use of coatings. For graphite or noncodified materials, this information should include the material grade or specification and the plan for developing the relevant material properties. For composites, the composite fiber, matrix material, architecture, manufacturing method, and necessary material properties should be identified. For some SSCs of higher safety significance or degradation susceptibility tied to construction, the applicant should provide additional information related to fabrication, joining, and/or postprocessing/heat treatment to be provided on the docket. Generally, highly detailed supporting documentation related to materials should be available for audit but does not require docketing. <u>Graded Approach:</u> Information on materials of construction should be provided commensurate with the safety significance of the SSC. For SSCs in less harsh environments (e.g., lower temperature, inert chemistry, low irradiation levels) or of lesser safety significance (e.g., some NSRST SSCs), the staff can generally accept higher level information on materials of construction at the CP stage, based on the lower likelihood of materials degradation or lower safety significance of the SSC. For example, rather than a specific alloy or grade of material, in these situations an applicant could identify a class of materials (e.g., austenitic stainless steels) or a short list of options, with the commitment to use a compatible weld metal.	In general, for NST SSCs, applicants should provide information related to their potential to interfere with the ability of SR and NSRST SSCs to perform their safety functions. Little information is needed beyond that.
Identification of environmental conditions	The CP application should include preliminary or bounding estimates for the following: • temperature • coolant (or other) chemistry • SSC design lifetime • stress/loading • if applicable, irradiation fluence (neutron or gamma depending on the material) These environmental conditions should be provided for normal operations as well as off-normal/accident conditions (most relevant for temperature, stress, and chemistry). Detailed supporting documentation for the estimates of environmental conditions should be available for audit but generally would not need to be docketed at the preliminary design stage. <u>Graded Approach:</u> Information on environmental conditions should be provided commensurate with the safety significance of the SSC and the likelihood of degradation. For SSCs in less harsh	

Key Area	Safety-Significant (SR or NSRST) SSCs	NST SSCs
	environments (e.g., lower temperature, inert chemistry, lower irradiation levels) or of lesser safety significance (e.g., some NSRST SSCs), the staff can generally accept higher level information on environmental conditions at the CP stage, based on the lower likelihood of materials degradation or lower safety significance of the SSC.	
Use of codes and standards	The CP application should identify C&S for design, construction, and operation. If using endorsed nuclear C&S, limited additional information on C&S is needed beyond establishing how the preliminary design information is consistent with the C&S. If using non-endorsed C&S, the application should also describe special treatments to be applied in design, fabrication, and construction, with justification for why the C&S and special treatments are appropriate given the risk and safety significance of the SSC, and why they are consistent with relevant PDC or FDC. Preliminary design and fabrication specifications for safety-significant SSCs should be available for audit. <u>Graded Approach:</u> Generally, NRC guidance indicates that endorsed nuclear C&S should be used for SR SSCs, while it is acceptable to use non-endorsed C&S with appropriate special treatments for NSRST SSCs. At the CP stage, the applicant should identify C&S for all SR SSCs and important or major NSRST SSCs, consistent with the consequences and likelihood of failure.	
Environmental compatibility assessment	Consistent with 10 CFR 50.34(a)(3)(iii) and 10 CFR 53.1309, the CP application should provide a plan to address environmental compatibility and ensure materials performance for safety-significant SSCs. The plan should describe how adequate passive SSC performance is ensured, considering the safety function, material, and operating environment for each safety-significant SSC and potential degradation mechanism. When appropriate, the CP application should identify research and development (R&D) activities associated with ensuring materials performance as an R&D program consistent with 10 CFR 50.34(a)(8) or 10 CFR 53.1309(a)(2)(iii). Acceptable materials performance for each potential degradation mechanism is typically demonstrated using a combination of the following types of information: • existing or historical experimental data • new experimental data generated by ongoing or planned testing • operating experience • analysis • performance monitoring Consistent with a preliminary design at the CP stage, the application should contain a preliminary assessment of how degradation mechanisms will be managed to ensure adequate materials performance. This may include preliminary analysis along with a description of planned testing and	

Key Area	Safety-Significant (SR or NSRST) SSCs	NST SSCs
	analysis for the final design that it may be appropriate to identify as R&D. Plans for performance monitoring may be an effective way to demonstrate acceptable materials performance in the absence of sufficient relevant (or robust) upfront experimental data or operational experience. <u>Graded Approach:</u> Information on environmental compatibility and materials performance should be provided commensurate with the safety significance of the SSC. SR SSCs generally have higher safety significance than NSRST SSCs. NSRST SSCs are safety significant but typically less so than SR SSCs, and applicants should demonstrate reasonable confidence of performance of safety functions consistent with their reliability and capability targets (if applicable). Consistent with the generally lower safety significance of NSRST SSCs, the depth and scope of review will generally be more limited for NSRST SSCs than for SSCs with higher safety significance, so less information will generally need to be docketed.	
Performance monitoring	The CP application should include commitments to design safety-significant SSCs to be inspectable or able to be effectively monitored by surveillance coupons or other means, to the extent that these are determined to be necessary to demonstrate conformance with relevant design criteria. Detailed performance monitoring information (e.g., exact scope, inspection frequency) need not be provided at the CP stage. Supporting documentation should be available for audit that describes the feasibility of potential performance monitoring methods and how the design provides access for performance monitoring. Applicants should be prepared to docket appropriate information as necessary to support the findings needed for the NRC safety evaluation. <u>Graded Approach:</u> Information related to performance monitoring should be commensurate with the safety significance of the SSC. SR SSCs generally will have more PDC or FDC-driven requirements for inspection and monitoring than NSRST SSCs. NSRST SSCs typically have fewer PDC or FDC-related requirements but should undergo appropriate performance monitoring to ensure that they will perform their safety functions sufficiently, consistent with their reliability and capability targets (if applicable).	

Table 2 identifies the information that should be provided at the final design stage in an OL, COL, SDA, DC, or ML application consistent with the requirements of 10 CFR 50.34(b), 10 CFR 52.47, 10 CFR 52.79, 10 CFR 52.137, 10 CFR 52.157, 10 CFR 53.1309, 10 CFR 53.1369, 10 CFR 53.1416, 10 CFR 53.1209, 10 CFR 53.1239, and 10 CFR 53.1279. The value and form of preapplication engagement described above for CPs also applies for OL, COL, SDA, DC, or ML applications.

Table 2 Application Content Information on Passive SSC Materials Performance that Should be Available During Reviews of Final Design

Key Area	Safety-Significant (SR or NSRST) SSCs	NST SSCs
Identification of materials of construction	The application should provide final detailed design information on materials of construction. For metallic SSCs, this information should include specific alloy or grade of material, material specification number, product form, weld metals, welding process, other joining methods (e.g., bolted flanges), and use of coatings or other post-fabrication processing. For nonmetallic or noncodified materials, this information should include the material grade or specification or composite fiber and matrix material and architecture, as well as final material properties used to perform a safety analysis. Highly detailed information that is generally included in design and fabrication specifications, such as final geometric tolerances and welding or post-processing details, should be available for audit but generally do not need to be included in the FSAR. For ML applications, however, post-fabrication information may need to be submitted on the docket, consistent with the 10 CFR 53.1279(b)(1) requirements to include information on the processes that will be used to procure, fabricate, and assemble components that make up the manufactured reactor. <u>Graded Approach:</u> Information on materials of construction should be provided commensurate with the safety significance of the SSC. For SSCs in less harsh environments (e.g., lower temperature, inert chemistry, lower irradiation levels) or of lesser safety significance (e.g., some NSRST SSCs), the applicants can provide less information on materials of construction in the FSAR, based on the lower likelihood of materials degradation or lower safety significance of the SSC. However, final detailed design information should be available for audit during the relevant review process, and applicants should be prepared to docket information as necessary to support the findings needed for the NRC safety evaluation.	In general, for NST SSCs, applicants should provide information related to limited their potential to interfere with the ability of SR and NSRST SSCs to perform their safety functions. Little information is needed beyond that.
Identification of environmental conditions	The application should include final information on environmental conditions such as the following: • temperature • coolant (or other) chemistry • SSC design lifetime • stress/loading • (if applicable) irradiation fluence These environmental conditions should be provided for normal operations as well as off-normal/accident conditions (most relevant for temperature, stress, and chemistry) and used as appropriate with the C&S applied to the design and construction of the SSC. These environmental	

Key Area	Safety-Significant (SR or NSRST) SSCs	NST SSCs
	conditions should also be used as inputs to the RIM process from XI-2, or in the development of a performance monitoring program outside of RIM. <u>Graded Approach:</u> Information on environmental conditions should be provided commensurate with the safety significance of the SSC and the likelihood of degradation. For SSCs in less harsh environments (e.g., lower temperature, inert chemistry, lower irradiation levels) or of lesser safety significance (e.g., some NSRST SSCs), applicants can provide less information on environmental conditions in the FSAR, based on the lower likelihood of materials degradation or lower safety significance of the SSC. However, final detailed information on environmental conditions should be available for audit during the review process, and applicants should be prepared to docket information as necessary to support the findings needed for the NRC safety evaluation.	
Use of codes and standards	The application should include final identification of C&S (and special treatments if applicable) for design, construction, and operation along with the detailed final design information to confirm consistency with the specified C&S. Highly detailed information such as design calculations and design specifications should be available for audit, to allow the staff to confirm on a sampling basis how the endorsed C&S are being satisfied. Selected aspects of this information may need to be docketed to support specific NRC safety evaluation findings. <u>Graded Approach:</u> For SSCs using endorsed nuclear C&S, the staff review is focused on confirming consistency of the final design with the specified C&S, since the adequacy of the C&S has already been established. For SSCs using non-endorsed C&S, the staff review includes confirming the adequacy of the specified C&S and special treatments given the risk and safety significance of the SSC, as well as confirming consistency of the final design with the specified C&S and special treatments. However, given that the non-endorsed C&S will generally be used for SSCs with lower safety significance, the depth and scope of review will be more limited for these SSCs than for SSCs with higher safety significance, so less detail is likely to be needed on the docket.	
Environmental compatibility assessment	Consistent with the content of application requirements identified above, the application should provide final justification for materials performance to qualify and address environmental compatibility for safety-significant SSCs. The justification based on the final detailed design information should address how adequate passive SSC performance is ensured, considering the safety function, material, and operating environment for each safety-significant SSC and potential degradation mechanism. Acceptable materials performance for each potential degradation mechanism is typically demonstrated using a combination of the following types of information:	

Key Area	Safety-Significant (SR or NSRST) SSCs	NST SSCs
	- existing or historical experimental data - new experimental data generated by ongoing or planned testing - operating experience - analysis - performance monitoring In the absence of sufficient relevant (or robust) upfront experimental data or operational experience, an important means of ensuring acceptable materials performance is to use performance monitoring methods with demonstrated effectiveness to identify SSC degradation before safety issues arise. A particular challenge is the management of degradation due to combined environmental effects not easily captured in experimental testing, such as combinations of the effects of high temperature (e.g., thermal aging embrittlement), corrosive environments (e.g., dealloying mechanisms), and irradiation (e.g., irradiation embrittlement). Surveillance coupons or other performance monitoring may be an effective way to improve understanding of combined environmental effects on material properties. It may also be appropriate to use conservative assumptions for degraded properties based on experimental testing and analysis. <u>Graded Approach:</u> Information on environmental compatibility and materials performance should be provided commensurate with the safety significance of the SSC. SR SSCs generally have higher safety significance than NSRST SSCs. NSRST SSCs are safety significant but typically less so than SR SSCs, and applicants should demonstrate reasonable confidence of performance of safety functions consistent with their reliability and capability targets (if applicable). Consistent with the generally lower safety significance of NSRST SSCs, the depth and scope of review will generally be more limited for NSRST SSCs than for SSCs with higher safety significance, so less information will generally be docketed.	
Performance monitoring	Consistent with the content of application requirements identified above, the application should include final detailed information on performance monitoring, such as in-service inspection, surveillance coupons, and continuous monitoring that may be performed to confirm passive SSC performance. Final detailed information on performance monitoring information should include the following: - inspection methods, scope, and frequency - surveillance coupon program information, including the methodology for collecting and analyzing surveillance coupon data to demonstrate safe operation	

Key Area	Safety-Significant (SR or NSRST) SSCs	NST SSCs
	description of parameters subject to continuous monitoring, such as online monitoring for cracking, pressure, temperature, or chemistry changes to indicate a leak or loss of boundary integrity, along with an explanation of how this monitoring is tied to passive SSC performance Highly detailed information such as the basis for development of reliability and capability targets (if applicable) and performance demonstration for inspection or monitoring methods should be available for audit, to allow the staff to confirm the effectiveness of the performance monitoring approach. Selected aspects of this information may need to be docketed to support specific NRC safety evaluation findings. <u>Graded Approach:</u> Information on environmental compatibility and materials performance should be provided commensurate with the safety significance of the SSC. SR SSCs generally will have more requirements for inspection and monitoring through specific PDC or FDC than NSRST SSCs. Consistent with these differences in requirements and safety significance, the depth and scope of review will generally be more limited for NSRST SSCs than for SSCs with higher safety significance, so less information will generally need to be docketed. The staff should also give appropriate consideration to any unique deployment concepts applicable to the design, such as a short operating life relative to operating LWRs, and how they may affect the appropriate scope of performance monitoring.	

APPENDIX B

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