U.S. NUCLEAR REGULATORY COMMISSION

REGULATORY GUIDE 1.87, REVISION 3

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ACCEPTABILITY OF ASME CODE, SECTION III, DIVISION 5, "HIGH TEMPERATURE REACTORS"

A. INTRODUCTION

Purpose

This regulatory guide (RG) describes an approach that is acceptable to the staff of the U.S. Nuclear Regulatory Commission (NRC) to assure the mechanical/structural integrity of components that operate in elevated temperature environments and that are subject to time-dependent material properties and failure modes. It endorses, with exceptions and limitations, the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) Code (ASME Code) Section III, "Rules for Construction of Nuclear Facility Components," Division 5, "High Temperature Reactors" (Ref. 1), and several related Code Cases.

Applicability

This RG applies to non-light-water reactor (non-LWR) licensees and applicants subject to Title 10 of the Code of Federal Regulations (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities" (Ref. 2), and 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants" (Ref. 3).

Applicable Regulations

- 10 CFR Part 50 provides regulations for licensing production and utilization facilities.
    - 10 CFR 50.34(a)(4) requires applicants to include in the preliminary safety analysis report a preliminary analysis and evaluation of the design and performance of structures, systems, and components (SSCs) of the facility with the objective of assessing both the risk to public health and safety from facility operation, including determining the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and the adequacy of SSCs provided to prevent accidents and mitigate their consequences.

Written suggestions regarding this guide may be submitted through the NRC's public website in the NRC Library at https://www.nrc.gov/reading-rm/doc-collections/reg-guides/index.html., under Document Collections, in Regulatory Guides, at https://www.nrc.gov/reading-rm/doc-collections/reg-guides/contactus.html, and will be considered in future updates and enhancements to the "Regulatory Guide" series. During the development process of new guides suggestions should be submitted within the comment period for immediate consideration. Suggestions received outside of the comment period will be considered if practical to do so or may be considered for future updates.

Electronic copies of this RG, previous versions of RGs, and other recently issued guides are also available through the NRC's public website in the NRC Library at https://www.nrc.gov/reading-rm/doc-collections/reg-guides/index.html under Document Collections, in Regulatory Guides. This RG is also available through the NRC's Agencywide Documents Access and Management System (ADAMS) at https://www.nrc.gov/reading-rm/adams.html, under ADAMS Accession Number (No.) ML25176A084. The regulatory analysis is associated with non-rulemaking and may be found in ADAMS under Accession No. ML24275A266. The associated draft guide DG-1436, Revision 3, may be found in ADAMS under Accession No. ML24275A266, and the staff responses to the public comments on DG-1436 may be found under ADAMS Accession No. ML25188A048.

- 10 CFR 50.34(b)(6)(iv) requires an application for an operating license to include in the final safety analysis report plans for conducting normal operations, including maintenance, surveillance, and periodic testing of SSCs.

- 10 CFR Part 50, Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,” establishes quality assurance requirements for the design, manufacture, construction, and operation of those SSCs that prevent or mitigate the consequences of postulated accidents that could cause undue risk to public health and safety.

- 10 CFR Part 52 governs the issuance of early site permits, standard design certifications, combined licenses, standard design approvals, and manufacturing licenses for nuclear power facilities.

  - 10 CFR 52.79(a)(5), in part, requires an application for a combined license to include an analysis and evaluation of the design and performance of SSCs with the objective of assessing both the risk to public health and safety from facility operation, including determining the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and the adequacy of SSCs intended to prevent accidents and mitigate the consequences of accidents.[1]

  - 10 CFR 52.79(a)(29), in part, requires an application for a combined license to include plans for conducting normal operations, including maintenance, surveillance, and periodic testing of SSCs.

Related Guidance

- NUREG-2245, “Technical Review of the 2017 Edition of ASME Section III, Division 5, “High Temperature Reactors”” (Ref. 4), documents the NRC staff’s review of the 2017 Edition of ASME Code Section III, Division 5 and Code Cases N-861, N-862. Exceptions and limitations remaining from 1.87, Rev 2 in this version of the RG still use this NUREG as documentation and will be noted below.

- Appendix A, “General Design Criteria for Nuclear Power Plants,” to 10 CFR Part 50 contains the general design criteria (GDC), which establish the 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 been issued by the Commission. Appendix A also indicates that the GDC are generally applicable to other types of nuclear power units and are intended to provide guidance in determining the principal design criteria for such other units.

  - 10 CFR Part 50, Appendix A, GDC 1, “Quality Standards and Records,” requires, in part, that SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed. Where generally recognized codes and standards are used, GDC 1 provides that they be identified and evaluated to determine their applicability, adequacy, and sufficiency and be supplemented or modified as necessary to ensure a quality product in keeping with the required safety function.

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[1] Similar requirements for design certifications, standard design approvals, and manufacturing licenses are detailed in other subparts of 10 CFR Part 52.

- RG 1.232, "Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors," (Ref. 5), lists the NRC's proposed guidance on how the GDC in 10 CFR Part 50, Appendix A, may be adapted for non-LWR designs.

    - Advanced reactor design criterion (ARDC) 1, "Quality Standards and Records," in RG 1.232, provides one principal design criterion to the effect, in part, that SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed. Where generally recognized codes and standards are used, ARDC 1 provides that they be identified and evaluated to determine their applicability, adequacy, and sufficiency and be supplemented or modified as necessary to ensure a quality product in keeping with the required safety function.

- RG 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants" (Ref. 6), describes a quality classification system for components containing water, steam, or radioactive material in light-water-cooled nuclear power plants.

- DANU-ISG-2023-1 "Material Compatibility for non-Light Water Reactors Interim Staff Guidance" (Ref. 7), provides guidance to assist NRC staff in reviewing applications for construction and operation of non-light water reactor designs.

Purpose of Regulatory Guides

The NRC issues RGs to describe methods that are acceptable to the staff for implementing specific parts of the agency's regulations, to explain techniques that the staff uses in evaluating specific issues or postulated events, and to describe information that the staff needs in its review of applications for permits and licenses. Regulatory guides are not NRC regulations and compliance with them is not required. Methods and solutions that differ from those set forth in RGs are acceptable if the applicant provides sufficient basis and information for the NRC staff to verify that the alternative methods comply with the applicable NRC regulations.

Paperwork Reduction Act

This RG provides voluntary guidance for implementing the mandatory information collections in 10 CFR 50.55a, 10 CFR Part 50 and 10 CFR Part 52 that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.). These information collections were approved by the Office of Management and Budget (OMB), under control numbers 3150-0264, 3150-0011 and 3150-0151, respectively. Send comments regarding this information collection to the FOIA, Library, and Information Collections Branch (T6-A10M), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by e-mail to Infocollects.Resource@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0264, 3150-0011 and 3150-0151), Office of Management and Budget, 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.

B. DISCUSSION

Reason for Revision

Revision 2 of this RG endorsed the 2017 Edition of ASME Code Section III, Division 5 and Code Cases N-872 and N-898, which permit the use of nickel-based Alloy 617. This revision (Revision 3) updates the guidance to endorse, with exceptions and limitations, the 2023 Edition of ASME Code Section III, Division 5, as a method acceptable to the staff for the materials, mechanical/structural design, construction, testing, and quality assurance of mechanical systems and components and their supports in high-temperature reactors. This revision removes conditions from Revision 2 of the RG that have been addressed in the 2023 version of the Code. This revision also endorses, with exceptions and limitations, the Code Cases N-812-1, N-861-2, N-862-2, N-872, N-898-1, N-924 and N-940.

Background

ASME Code, Section III, Division 1, "Rules for Construction of Nuclear Power Plant Components" (Ref. 8), contains the rules of construction of ASME Class 1, 2, 3, metal containment components and their supports, and core support structures. These rules generally apply to time-independent material strength and deformation, with a maximum allowable temperature of 370 degrees Celsius (°C) (700 degrees Fahrenheit [°F]) for some materials and 425 °C (800 °F) for others. The NRC incorporates by reference portions of the ASME Code, Section III, Division 1, in 10 CFR 50.55a.

Some new reactor designs would operate at temperatures above the limits specified in ASME Code, Section III, Division 1. ASME Code Section III, Division 5 extends the provisions of ASME Code, Section III, Division 1, to allow the construction of metallic nuclear plant components that would operate within the material strength and deformation time-dependent regime (creep regime), and address elevated temperature[2] conditions (e.g., temperatures greater than 370 °C [700 °F] for low-alloy steels or 425 °C [800 °F] for austenitic stainless steels). In addition, ASME Code, Section III, Division 5, provides new provisions for the construction of certain nuclear plant components using graphite and composite materials.

The NRC issued RG 1.87, Revision 2 (Ref. 9), NUREG-2245, and Technical Letter Report (TLR) RES/DE/REB-2022-1 (Ref. 10) to endorse, with exceptions and limitations, and provide technical bases for the endorsement of the 2017 edition of ASME Code, Section III, Division 5 "High Temperature Reactors" and Code Cases N-872 and N-898, which permit the use of Nickel-Based Alloy 617. Certain exceptions and limitations in Revision 2 remain in this revision and will be noted. For exceptions and limitations on changes made between the 2017 and 2023 edition of ASME Code, Section III, Division 5, the basis for the regulatory positions can be found in "Bases for NRC Staff Regulatory Guidance Positions" section in this document.

Section C of this RG lists the exceptions to and limitations on the NRC staff's endorsement of ASME Code, Section III, Division 5, including the endorsement of Code Cases, N-812-1, N-861-2, N-862-2, N-872, N-898-1, N-924 and N-940.

Appendix A to this RG provides guidance for the quality group classification of components in non-LWR designs. In addition, it provides one method that is acceptable to the NRC staff for the selection of quality standards with respect to the safety classification of components for non-LWR nuclear power plants. An

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2 ASME Code, Section III, Division 5, defines elevated temperature as temperature in excess of those temperatures established in Table HAA-1130-1, "Values of T_{max} for Various Classes of Permitted Materials."

applicant or licensee may request the use of a classification system for components in its non-LWR nuclear power plant as an alternative to that described in Appendix A to this RG.

Bases for NRC Staff Regulatory Guidance Positions

The following items discuss the bases for the NRC staff's positions stated in Section C of this RG on potential issues or concerns when implementing the 2023 ASME Code, Section III, Division 5. The bases of staff positions that remain from Revision 2 of this RG, can be found in NUREG-2245 and RES/DE/REB-2022-1.

Basis for Regulatory Guidance Position a (1)

The term "items commensurate with their contribution to safety or risk" is vague and there is a need to clarify which SSCs are appropriate to use in these sections of code. The final portion of this limitation indicates the staff's openness to applying these alternate rules to nonsafety-related with special treatment (NSRST) SSCs, while providing the staff flexibility to confirm these rules are appropriate based on the risk significance and reliability and capability targets of the NSRST SSCs for a specific design.

Basis for Regulatory Guidance Position i (1)

The stress values in Table HBB-I-14.2, Table HBB-I-14.3, Table HBB-I-14.4 and Table HBB-I-14.6 were established using data from non-welded products. Thus, applicants and licensees should provide justification when using the stress values in these tables for welded products. The NRC notes that, in some non-nuclear codes and standards, the allowable stress values of welded products are established by applying a reduction factor to the allowable stress values of non-welded products.

Basis for Regulatory Guidance Position m

Nonmandatory Appendix HBB-Y does not provide guidance to applicants or licensees relevant to submittals to the NRC. Additionally, it is not within the NRC's regulatory authority to endorse a procedure as acceptable for submitting materials to and meeting requirements of a third-party entity such as the ASME.

Basis for Regulatory Guidance Position n

Consistent with HBB-3214.2, when strain and stress histories from inelastic analyses are used to evaluate strain limits and creep-fatigue damage, they should be sufficiently comprehensive to predict significant behavioral features which include, but are not limited to, creep behavior and cyclic hardening/softening. Modeling of these behaviors as determined from experimental measurements is necessary, irrespective of whether they are the results of dislocation re-arrangements, evolution of material microstructure, or creep cavitation. This is consistent with the use of experimentally determined strain and stress histories, which are the macroscopic manifestation of these mechanisms which include cyclic hardening/softening and tertiary creep, to evaluate creep damage and fatigue damage per code provisions to establish the creep-fatigue interaction diagram.

Basis for Regulatory Guidance Position o

The Appendix HBB-Z inelastic model for the 316H stainless steel is based on the classical Chaboche viscoplastic formulation with two back stresses that evolve according to the competition between the plastic hardening and the dynamic recovery. It is augmented by a damage mechanics formulation applied to both elastic and inelastic deformation.

The damage parameter, ω, treated as an internal variable within the constitutive model, evolves as a function of the von Mises stress. It is always increasing except when the stresses are zero. The inclusion of the damage mechanics formulation allows the modeling of accelerated creep rate due to tertiary creep which is not captured by the specific baseline Chaboche model adopted. But when applied to the elastic deformation, it renders the elastic deformation time-history dependent and effectively reduces the elastic modulus continuously. This type of elastic behavior does not occur in metal. The strain and stress histories from this model shall not be used to evaluate strain limits and creep-fatigue damage when the internal variable ω exceeds 0.15.

Basis for Regulatory Guidance Position p

The inelastic model for 9Cr–1Mo–V is based on a formulation that distinguishes between rate-independent and rate-dependent responses spatially and temporally in the computational domain by using a Kocks-Mecking criterion based on the total strain rate. The tension/compression asymmetry behavior of 9Cr–1Mo–V is modeled by adding a pressure term to the classical Chaboche formulation of the flow function. The evolutions of the back stresses in the adopted baseline Chaboche formulation involve plastic hardening, dynamic recovery and static recovery. This baseline Chaboche formulation thus has the repositories for modeling primary, secondary, and tertiary creep, without the damage mechanics augmentation.

In the rate-dependent regime, the viscoplastic deformation is pressure-sensitive due to the pressure term in the flow function. However, the evolutions of the deviatoric back stresses are driven by both deviatoric and hydrostatic tensorial quantities. This causes the back stresses to drift away from the deviatoric stress space, leading to contradictions. For example, under uniaxial tension, the model gives contradictory evolution equations for the axial back stress when the response in the axial and transverse directions are considered.

In the rate-independent regime, the constitutive model is based on the standard rate-independent plasticity theory. However, the same flow function and evolution equations for the rate-dependent regime are employed for the rate-independent regime. Thus, in addition to the same issue on the deviatoric character of the back stresses during deformation, the rate-dependent nature of the back stress evolutions renders the equations not admissible for a rate-independent model.

Due to the complexity of the model formulation, it is not possible to identify a restricted design envelope where the contradictions in the model formulation would still lead to structural responses that could be deemed adequate.

Basis for Regulatory Guidance Position w

The NRC recognizes that composite materials are often designed for a specific application and the material behavior for a composite can be tailored to the specific needs of that application. The Designer is provided significant latitude within many aspects of HHB to address these needs. As there are not quantitative or qualitative requirements for many aspects which are assigned to the Designer, the NRC will need to review the decisions made by the Designer.

A short noncomprehensive list which is intended to provide examples of the things which will need to be reviewed by the NRC is included below.

1) Damage tolerance (HHB-3110)
2) Design critical stress in compressive loading (HHB-3145)
3) Stress analysis boundary conditions (HHB-3215)
4) Irradiated stress analysis (HHB-3215.3)
5) Material acceptance criteria (HHB-5332)
6) Material testing and data in the Material Data Sheet (HHB-II-1000(d) and HHB-II-1000(e))
7) Purity limits and degradation limits (HHB-I-1120 (f) and HHB-I-1120 (g))

Basis for Regulatory Guidance Position x

The temperature field applied to a Composite Core Component can induce loads independent of the effects of fast flux irradiation. It is more generally applicable to select the Design Temperature as the temperature field in combination with all other Design Loadings that results in the highest use of the Composite Core Component, instead of only in combination with the Design Fast Flux.

Basis for Regulatory Guidance Position y

The weight change limit of 1% which distinguishes between oxidized and non-oxidized components does not appear to be well supported in the literature. Literature has shown that, for SiC-SiC composites, significant strength losses have been seen at less than 1% mass loss (Ref. 12). There is not an apparent basis for why 1% is currently used at the weight change limit to designate oxidation.

Removing the low strength region from stress evaluations is conservative for load-controlled stresses but may not be conservative for thermal stresses. There may be additional stresses from secondary loads that are lost by removing the low strength region from the analysis.

The technical justification for removing material which has experienced more than 30% weight loss from the strength and chemical attack calculations is unclear. The weight loss limit at which the material should be removed from the calculations should be justified by the applicant.

HHB-3141(d), Alternative Methodology, prohibits combined chemical attack to weight loss greater than 1% occurring simultaneously with irradiation greater than 0.25 dpa. While these conditions are outside the scope of the code requirements, Composite Core Components subject to these conditions may still be able to perform their function. The Designer should develop an alternative methodology that justifies the acceptability of such a use.

Basis for Regulatory Guidance Position z

For C-C composites, experimental data suggest that irradiation effects occur below the 0.25 dpa limit set for classifying a C-C composite as nonirradiated. This is acknowledged within HHB-3215.2 which states that irradiation effects on thermal conductivity must be accounted for at doses above 0.001 dpa.

For SiC-SiC composites, properties including thermal conductivity, flexural strength, interfacial debond shear strength, and swelling change with irradiation and stabilize around 1 dpa, but the rules classifying SiC-SiC components as nonirradiated below 1.0 dpa disregard the transitional phase.

Basis for Regulatory Guidance Position aa

The limit of 100 meters per second mean gas flow velocity for evaluating the effects of erosion was taken from the rules on graphite, HHA-3143, and lacks technical justification for use with Composite

Core Components. Additionally, fission reactors may operate with fluid flows other than gas. The flow rate limit for evaluating erosion of composites should be justified by the applicant.

Basis for Regulatory Guidance Position bb

This paragraph reiterates the simplified assessment methodology which is conditioned.

Basis for Regulatory Guidance Position cc

This subparagraph is related to the simplified assessment methodology in HHB-3220, which is conditioned.

Basis for Regulatory Guidance Position dd

This subparagraph is related to the simplified assessment methodology in HHB-3220, which is conditioned.

Basis for Regulatory Guidance Position ee

HHB-3220 contains the simplified assessment for design by analysis in which the Maximum Loading Mode Stress is compared to an allowable stress value that depends on the target probability of failure. The lack of requirements in the stress analysis combined with a lack of validation for the analysis methodology has precluded the NRC from verifying the conservatism of the assessment. Additionally, there is insufficient guidance for determining the mode of failure and calculating the Maximum Loading Mode Stress.

When performing the design by analysis, the applicant should develop its own evaluation procedure and corresponding limits to address stress-time-temperature, irradiation, oxidation and other degradation effects. Justification for this procedure and the corresponding limits is required.

Basis for Regulatory Guidance Position ff

The code does not provide sufficient guidance to relate anisotropic strength parameters. It is not supported that, for example, the flexure strength measured in a given direction is proportionally related to the compressive strength measured in the same direction for composite materials.

Basis for Regulatory Guidance Position gg

Nonmandatory Appendix HHB-B does not provide guidance for applicants or licensees.

Basis for Regulatory Guidance Position hh

Nonmandatory Appendix HHB-C does not provide guidance for applicants or licensees.

Basis for Regulatory Guidance Position ii

Nonmandatory Appendix HHB-D does not provide guidance for applicants or licensees.

Basis for Regulatory Guidance Position jj

Nonmandatory Appendix HHB-E does not provide guidance for applicants or licensees.

Basis for Regulatory Guidance Position 1-CC-N-898-1

Consistent with HBB-3214.2, when strain and stress histories from inelastic analyses are used to evaluate strain limits and creep-fatigue damage, they should be sufficiently comprehensive to predict significant behavioral features which include, but are not limited to, creep behavior and cyclic hardening/softening. Modeling of these behaviors as determined from experimental measurements is necessary, irrespective of whether they are the results of dislocation re-arrangements, evolution of material microstructure, or creep cavitation. This is consistent with the use of experimentally determined strain and stress histories, which are the macroscopic manifestation of these mechanisms which include cyclic hardening/softening and tertiary creep, to evaluate creep damage and fatigue damage per code provisions to establish the creep-fatigue interaction diagram.

Basis for Regulatory Guidance Position 2-CC-N-898-1

The inelastic model for Alloy 617 is based on a formulation that distinguishes between rate-independent and rate-dependent responses spatially and temporally in the computational domain by using a Kocks-Mecking criterion based on the total strain rate.

In the rate-dependent regime, the inelastic model is formulated using the classical Chaboche framework where the back stresses are evolving in a rate-independent manner through the competition between a plastic hardening term and a dynamic recovery term. This baseline Chaboche framework only has repositories for primary creep and secondary creep. The baseline Chaboche equations are augmented by a continuum damage mechanics framework applied to both elastic and inelastic deformation. The damage parameter, treated as an internal variable within the constitutive model, evolves as a function of the von Mises stress. Hence it is always increasing except when the stresses are zero. The damage parameter is explicitly introduced into the flow function to allow for the modeling of accelerated creep rate due to tertiary creep.

In the rate-independent regime, the constitutive model is based on the standard rate-independent plasticity theory. However, the same flow function, the same evolution equations for the internal variables, and the same continuum damage mechanics augmentation as the rate-dependent case are employed for the rate-independent regime. However, since the evolution equation for the damage parameter is time-dependent, this leads to a rate-dependence in the flow factor when the plasticity consistency condition is enforced. This is not admissible in the standard rate-independent plasticity theory.

Since the damage parameter is applied to elastic and inelastic deformation in both rate-dependent and rate-independent regimes, the constitutive equations lead to time-history dependent elastic deformation and effectively reduce the elastic modulus continuously. This type of elastic behavior does not occur in metal.

It is noted that the rate-independent and rate-dependent responses can coexist in different regions of a structural component. The issues associated with the rate-independent equations could lead to inaccurate stress and strain predictions in the rate-dependent part of the component since solutions in both regimes need to be solved for simultaneously.

Due to the complexity of the model formulation, it is not possible to identify a restricted design envelope where the inadmissible equations in the model would still give rise to structural responses that could be deemed adequate.

Basis for Regulatory Guidance Position 1-CC-N-940

The use of encoded ultrasonic testing (UT) allows data analysis by additional qualified examiners and permits future examination results to be compared to original UT examination results. This process also provides a permanent record of the examination data. This is consistent with the requirements in ASME B31.1 (Ref. 15) paragraph 136.4.6(a)(1) and Code Case N-831-1 (Ref. 16) paragraph (g).

Basis for Regulatory Guidance Position 2-CC-N-940

This position provides qualification requirements to demonstrate the effectiveness of the UT examination systems, which is reasonable for safety-significant SSCs. Section V, Article 14 (Ref. 17) Intermediate Rigor requires limited performance demonstration. This is consistent with the requirement in ASME B31.1 paragraph 136.4.6(a)(2) that the procedures and equipment used to collect and analyze UT data need to be demonstrated.

Basis for Regulatory Guidance Position 3-CC-N-940

This position is consistent with industrial code ASME B31.1 Table 136.4.1-1, which requires 100% radiographic testing (RT) or UT for piping over NPS 2 that operates at temperatures over 750°F.

Basis for Regulatory Guidance Position 4-CC-N-940

The initial sample requirement in commercial codes (e.g., B31.1 and B31.3) for SSCs that are most analogous to those for nuclear service within the scope of this Code Case generally varies between 5% and 100%, depending on their design of the component and service condition. Given this variance and without specific information about the SSC to which the Code Case may be applied, staff cannot conclude that a 5% sample is generically appropriate. Position (1) in condition 4-CC-N-940 in Table 5 is intended to provide a baseline on inspection sampling to provide reasonable confidence of performance, while meaningfully lowering from the 100% inspection required for SR SSCs under regular ASME BPVC Section III rules and under commercial codes in certain situations. Position (2) provides explicit clarity that a CC user may propose what it believes to be an appropriate sample percentage based on the specific design and service condition of the SSC. Meanwhile, position (3) provides a simpler approach that does not require additional analysis or justification and is more likely to be applied in cases where there are fewer welds to inspect. Consistent with a graded approach, NRC staff finds 50% random sampling to be an acceptable approach based on the generally lower safety significance of NSRST SSCs relative to SR SSCs.

Basis for Regulatory Guidance Position 5-CC-N-940

Given the uncertainty of the characteristics of fluid systems other than water, the condition in 5-CC-N-940 is needed to ensure the definition of moderate energy piping is appropriately considered for each technology.

Basis for Regulatory Guidance Position 6-CC-N-940

The condition in 6-CC-N-940 is consistent with industrial code ASME B31.1, which only allows initial service leak tests instead of pressure test “when other types of tests are not practical or when leak tightness is demonstrable due to the nature of the service.”

Consideration of International Standards

The International Atomic Energy Agency (IAEA) works with member states and other partners to promote the safe, secure, and peaceful use of nuclear technologies. The IAEA develops Safety Requirements and Safety Guides for protecting people and the environment from harmful effects of ionizing radiation. This system of safety fundamentals, safety requirements, safety guides, and other relevant reports, reflects an international perspective on what constitutes a high level of safety. To inform its development of this RG, the NRC has considered IAEA Safety Requirements and Safety Guides pursuant to the Commission's International Policy Statement (Ref. 13) and Management Directive and Handbook 6.6, "Regulatory Guides" (Ref. 14).

The NRC staff considered the following IAEA Safety Standard in the update of the RG:

- International Atomic Energy Agency, "Safety of Nuclear Power Plants: Design, Specific Safety Requirements," IAEA Safety Standards Series No. SSR-2/1 (Rev. 1), (Ref. 19).

Documents Discussed in Staff Regulatory Guidance

This RG endorses, in part, the use of one or more codes or standards developed by external organizations, and other third-party guidance documents. These codes, standards and third-party guidance documents may contain references to other codes, standards or third-party guidance documents ("secondary references"). If a secondary reference has itself been incorporated by reference into NRC regulations as a requirement, then licensees and applicants must comply with that standard as set forth in the regulation. If the secondary reference has been endorsed in a RG as an acceptable approach for meeting an NRC requirement, then the standard constitutes a method acceptable to the NRC staff for meeting that regulatory requirement as described in the specific RG. If the secondary reference has neither been incorporated by reference into NRC regulations nor endorsed in a RG, then the secondary reference is neither a legally-binding requirement nor a "generic" NRC approved acceptable approach for meeting an NRC requirement. However, licensees and applicants may consider and use the information in the secondary reference, if appropriately justified, consistent with current regulatory practice, and consistent with applicable NRC requirements.

C. STAFF REGULATORY GUIDANCE

1. ASME Code, Section III, Division 5

The NRC staff endorses the 2023 Edition of the ASME Code, Section III, Division 5, as a method acceptable to the NRC staff for the materials, mechanical/structural design, construction, testing, and quality assurance of mechanical systems and components and their supports of high-temperature reactors, with the exceptions and limitations stated below. When Section III, Division 1 is referenced in this RG, the NRC staff is referring to the 2023 Edition**.** Where Division 5 references portions of Division 1, the NRC staff is endorsing use of those portions of Division 1 for high-temperature reactors, with the exceptions and limitations stated below. An applicant who wishes to follow the guidance in this RG should describe in its Final Safety Analysis Report or Quality Assurance Plan how the exceptions and limitations on the use of ASME Code, Section III, Division 5, will be addressed. The NRC staff endorses the 2023 Edition of the ASME Code, Section III, Division 5, with limitations, for application in the design and construction of high-temperature reactors, except where ASME identifies portions of the Code as being in the course of preparation. The NRC staff is unable to review those sections identified as being in the course of preparation to determine whether they are acceptable, and therefore, the staff does not endorse them.

a. When using ASME Code, Section III, Division 5, where Division 5 references ASME Code, Section III, Division 1, applicants and licensees should follow any applicable conditions for Division 1 that are identified in 10 CFR 50.55a.

    (1) The phrase "items commensurate with their contribution to safety or risk" appears in Code Case N-940, HBA-2610, HBB-2610 and within ASME Code, Section III, Division 1. In the application of ASME Code, Section III, Division 5 design rules, there is a need to clarify for which SSCs these rules are appropriate. The alternate requirements in these sections should not be applied to safety-related SSCs but may be appropriate for use for SSCs categorized as NSRST under RG 1.233 (Ref. 20). The justification for use of alternate requirements in these sections as a special treatment to achieve the reliability and capability targets specified for the NSRST SSC is subject to NRC review and approval. The NRC may review classification of NSRST SSCs in accordance with the approved methodology in RG 1.233.

b. HAA-1110, Scope; HAB-3255, Certification of the Design Specifications; HAB-3352, Design Report; HAB-3360, Certification of Construction Specification, Design Drawings, and Design Report; HAB-8161, Evaluation for a Certificate; HCB-3115, Design Report and Certification

    (1) The NRC staff does not endorse paragraph XXIII-1223 from Mandatory Appendix XXIII in ASME Code, Section III, "Appendices." When applying the 2017 and later editions of ASME Code Section III, the NRC does not endorse applicant and licensee use of a Certifying Engineer who is not a Registered Professional Engineer qualified in accordance with paragraph XXIII-1222 for Code-related activities that are applicable to NRC-regulated facilities.

c. Where ASME identifies portions of ASME Code, Section III, Division 5, as being in the course of preparation, the NRC staff is unable to review those sections to determine whether or not they are acceptable, and therefore, the staff does not endorse them.

d. HAB-3126, Subcontracted Calibration Services; HAB-3127, Subcontracted Testing Services; and HAB-4555.3, Approval and Control of Suppliers of Subcontracted Services

When using HAB-3126(a) and HAB-3127(a), and HAB-4555.3(c)(1) and (d)(1):

(1) The laboratory should be accredited based on an on-site accreditation assessment performed by the selected Accrediting Body within the past 48 months. The laboratory's accreditation should not be based on two consecutive remote accreditation assessments.

When using HAB-3126(b) HAB-3127(b), and HAB-4555.3(c)(2) and (d)(2):

(2) The procurement document should also specify that performance of the procured services[3] is contingent on the laboratory's accreditation being achieved through an on-site accreditation assessment by the Accreditation Body within the past 48 months.

e. HAB-5230, Scope of Work, Design Specifications, and Design Report

(1) When using HAB-5230, the applicant or licensee should also apply NCA-5230(d), when referencing the scope of work of the inspector.

f. HAB-5290, Data Reports and Construction Reports

(1) When using HAB-5290, the applicant or licensee should also apply NCA-5290(c)(1) and (c)(2), replacing the reference to NCA with HAB.

g. HBB-3600, Piping Design; HBB-3660, Design of Welds; HCB-3150, Limitations on Use; HCB-4000, Fabrication and Installation.

(1) The staff does not endorse the use of Division 5 provisions in accordance with HBB-3600, HBB-3660, HCB-3150, and HCB-4000 for socket welded fittings used in pressure-retaining joints and referenced in HBB-3000, HCB-3000 and HCB-4000, for welds with leg size less than 1.09* t_n, where t_n is the nominal pipe thickness.

h. HBB-6212(a), Test Medium and Test Temperature

(1) When using HBB-6212(a), the "nonhazardous liquid" should be (a) nonhazardous relative to possible reactions between residual test liquid and the normal coolant fluid and (b) nonhazardous with respect to deleterious effects to the component (material) (such as through corrosion by either the test liquid or a fluid created by reaction of test liquid and coolant).

(2) An applicant or licensee may justify a liquid as nonhazardous even if the liquid does not fall within the criteria in Item o.(1) above by employing post-test procedures that ensure proper draining and drying. When a test liquid is considered "nonhazardous" as a result of such prescribed post-test procedures, the post-test procedures should be documented and included as part of the appropriate Data Report Form specified by NCA-8400, as incorporated into Division 5 by HAA-1110(a).

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[3] For purposes of this RG, the term "service" is not used in its plain language meaning but rather used as defined in Subarticle HAB-9200, "Definitions," of ASME Code, Section III, Division 5 as follows. "Service: an activity performed by a subcontractor such as designing, machining, installation, repair, and nondestructive examination."

i. Mandatory Appendix HBB-I-14 Tables and Figures

(1) When using welded product forms listed in Table HBB-I-14.1(a), applicants and licensees should justify in the design report the use of the stress values in Tables HBB-I-14.2, HBB-I-14.3A, HBB-I-14.3B, HBB-I-14.3D, HBB-I-14.4A, HBB-I-14.4B, HBB-I-14.4D HBB-I-14.6A, HBB-I-14.6B, HBB-I-14.6D. Welded product forms include SA-249, SA-312, SA-358 and SA-403, Grade WP, Class W for Type 304 and 316 stainless steels, and SA-234 Grade WP22 welded fittings and SA-691, Grade 2¼ CR for 2¼Cr-1Mo steel.

(2) The NRC staff does not endorse the following materials properties in Mandatory Appendix HBB-I-14[4]. Note that interpolation between endorsed and unendorsed properties values is permissible only with appropriate justification.

(a) Type 304 stainless steel (Type 304 SS) values of S_{mt}, S_t, and S_r for the following time/temperature combinations (these are also shown graphically in Table 1):

1. US Customary Units
   a. Times greater than 30,000 hours at 1350 °F.
   b. Times greater than 3000 hours at 1400 °F.
   c. Times greater than 1000 hours at 1450 °F.
   d. Times greater than 100 hours at 1500 °F.

2. SI Units
   a. Times greater than 30,000 hours at 725 °C.
   b. Times greater than 3000 hours at 750 °C.
   c. Times greater than 1000 hours at 775 °C.
   d. Times greater than 300 hours at 800 °C.

[4] For all the S_{mt}, S_t, and S_r values not endorsed below, the temperature values are not exact conversions from US Customary to SI units for the same times. This is because ASME Code, Section III, Division 5, provides separate tables of allowable stresses (S_{mt}, S_t, and S_r) for US Customary and SI units in Appendix HBB-I-14, which are provided in increments of 50 °F for US Customary units, and 25 °C for SI units. The temperatures at which S_{mt}, S_t, and S_r were not endorsed were evaluated separately for the US Customary and SI tables. Use of either set of limitations is acceptable because any differences in allowable stresses resulting from conversion of temperatures and interpolation of allowable stresses are minor.

Table 1. Type 304 SS allowable stress limitations

(Gray shaded cells represent time/temperature combinations for which S_t, S_{mt}, and S_r, are not endorsed.)

US Customary Units											
Temp	Time (hr)										
°F	1	10	30	100	300	1k	3k	10k	30k	100k	300k
800											
850											
900											
950											
1000											
1050											
1100											
1150											
1200											
1250											
1300											
1350											
1400											
1450											
1500											

SI Units											
Temp	Time (hr)										
°C	1	10	30	100	300	1k	3k	10k	30k	100k	300k
425											
450											
475											
500											
525											
550											
575											
600											
625											
650											
675											
700											
725											
750											
775											
800											

(b) Type 316 stainless steel (Type 316 SS) S_r values for the following time/temperature combinations (these are also shown graphically in Table 2):

1. US Customary Units
   a. Times greater than 300 hours at 1400 °F.
   b. Times greater than 30 hours at 1450 °F.
   c. Times greater than 10 hours at 1500 °F.

2. SI Units
   a. Times greater than 300 hours at 750 °C.
   b. Times greater than 30 hours at 775 °C.
   c. Times greater than 30 hours at 800 °C.

Table 2. Type 316 SS limitations on S_r

(Gray shaded cells represent time/temperature combinations for which S_r is not endorsed.)

US Customary Units											
Temp	Time (hr)										
°F	1	10	30	100	300	1k	3k	10k	30k	100k	300k
800											
850											
900											
950											
1000											
1050											
1100											
1150											
1200											
1250											
1300											
1350											
1400											
1450											
1500											

SI Units											
Temp	Time (hr)										
°C	1	10	30	100	300	1k	3k	10k	30k	100k	300k
425											
450											
475											
500											
525											
550											
575											
600											
625											
650											
675											
700											
725											
750											
775											
800											

(c) 2-1/4Cr-1Mo material S_{mt}, S_t, and S_r values for the following time/temperature combinations (these are also shown graphically in Table 3)

1. US Customary Units
    a. Times greater than 100,000 hours at temperatures of 1000 °F and 1050 °F.
    b. Temperature greater than or equal to 1100 °F, for all times.

2. SI Units
    a. Times greater than 100,000 hours at temperatures of 525 °C and 550 °C.
    b. Temperature greater than or equal to 575 °C, for all times.

Table 3. 2-1/4Cr-1Mo allowable stress limitations

(Light Gray shaded cells represent time/temperature combinations for which S_{mt}, S_t, and S_r are not endorsed. Dark gray shaded cells are time/temperature combinations for which ASME Code, Section III, Division 5 does not provide allowable stress values)

US Customary Units											
	Time (hr)										
Temp °F	1	10	30	100	300	1k	3k	10k	30k	100k	300k
800											
850											
900											
950											
1000											
1050											
1100											
1150											
1200											
SI Units											
	Time (hr)										
Temp °C	1	10	30	100	300	1k	3k	10k	30k	100k	300k
425											
450											
475											
500											
525											
550											
575											
600											
625											
650											

(d) The R-factors in Tables HBB-I-14.10A-3 and HBB-I-14.10B-3 for Type 304 or Type 316 SS base metal welded with Type 316 SS filler using processes other than gas tungsten arc welding.[5]

(3) The NRC staff endorses Table HBB-I-14.1(a) with the following limitations:

(a) Note (2) to the table should be modified to add the following words: "The heat treatment is to be separately performed, and in-process heat treatment such as by direct quenching from hot forming is not permitted."

j. Nonmandatory Appendix HBB-T-1420, Limits Using Inelastic Analysis

(1) In applying the limits identified in HBB-T-1420 (including parameters such as strain, cycles, and temperature) in inelastic analysis, the applicants and licensees should validate the constitutive models used in assessments for cyclic creep loading. The validity of the inelastic constitutive models should be demonstrated in the design report.

k. Nonmandatory Appendix HBB-T-1510, General Requirements, and Nonmandatory Appendix HBB-T-1520, Buckling Limits

(1) When an applicant or licensee uses the strain factors in Table HBB-T-1521-1 for time-independent buckling, the applicant or licensee should justify in the design report that (1) the buckling is purely strain-controlled and not combined with load-controlled buckling and (2) significant elastic follow-up is not occurring.

l. Nonmandatory Appendix HBB-T-1710, Special Strain Requirements at Welds

(1) When using HBB-T-1710 applicants and licensees should develop their own plans to address the potential for stress relaxation cracking in their designs.

m. Nonmandatory Appendix HBB-Y, Guidelines for Design Data Needs for New Materials

(1) The NRC staff is not endorsing Nonmandatory Appendix HBB-Y because it is for information only. Additionally, it is not within the NRC's regulatory authority to endorse a procedure as acceptable for submitting materials to and meeting requirements of a third-party entity such as the ASME.

n. Nonmandatory Appendix HBB-Z-1212.3, Accumulated Damage

(1) The NRC staff is not endorsing this subparagraph.

o. Nonmandatory Appendix HBB-Z-1322, 316 SS

(1) The NRC staff endorses the constitutive model for 316H SS with the limitation that the strain and stress histories from the inelastic model shall only be used to evaluate strain limits and creep-fatigue damage when the internal variable ω does not exceed 0.15.

p. Nonmandatory Appendix HBB-Z-1325, 9Cr-1Mo-V

[5] Applicants wishing to use these base metal/weld metal combinations for welds made with processes other than gas tungsten arc welding may be able to demonstrate the adequacy of these R-factors by submitting additional data.

(1) The NRC staff is not endorsing this paragraph.

q. Mandatory Appendix HGB-III-2000, Buckling Limits: Time-Independent Buckling

(1) When an applicant or licensee uses the strain factors in Table HGB-III-2000-1 for time-independent buckling, the applicant or licensee should justify in the design report that (1) the buckling is purely strain-controlled and not combined with load-controlled buckling and (2) significant elastic follow-up is not occurring.

r. HHA-3142.4, Graphite Cohesive Life Limit

(1) The NRC staff is not endorsing the provisions of HHA-3142.4 that set the graphite cohesive life limit fluence to the fluence at which the material experiences a +10 percent linear dimensional change in the with-grain direction. Designers should determine the graphite cohesive life fluence limit beyond which the material is considered to provide no contribution to the structural performance of the Graphite Core Component (GCC) and justify that the limit is adequate for the GCC design.

s. HHA-3143, Abrasion and Erosion

(1) The NRC staff is not endorsing the provisions of HHA-3143 that set the mean gas flow velocity limit of 100 meters per second (330 feet per second) for evaluating the effects of erosion on the GCC design. Designers should determine the mean gas flow velocity limit above which an evaluation of erosion is necessary and justify that the limit is adequate for the GCC design.

t. HHA-4233.5, Repair of Defects and Flaws

(1) The NRC staff is not endorsing the provisions of HHA-4233.5 that set a maximum allowed repair depth of 2 millimeters (0.079 inch). Designers should determine a maximum allowed repair depth and justify that it is adequate for the GCC design, including consideration of the size of the component and the graphite grade(s) used.

u. Mandatory Appendix HHA-III-4200, Irradiated or Oxidized Graphite

(1) The NRC staff endorses HHA-III-4200 with the following exception: Irradiated or oxidized material property data used to populate the Material Data Sheet should come from testing performed on material that is representative of production billet specimens exposed to environmental conditions that are consistent with the qualification envelope defined in the Design Specification.

v. Nonmandatory Appendices HHA-A, Graphite as a Structural Material and HHA-B, Environmental Effects in Graphite

(1) The NRC staff is not endorsing Nonmandatory Appendices HHA-A and HHA-B because they are for information only and do not provide guidance for applicants or licensees.

w. HHB

(1) This is a general note on the entirety of HHB. In this subpart, the Designer is tasked with much of the technical development of the design requirements. Therefore, the NRC places the following condition: Any design decisions or determinations which are assigned to the Designer in HHB shall be subject to review by the NRC.

x. HHB-3123.2, Design Temperature Distribution

(1) The NRC staff is not endorsing the provisions of HHB-3123.2 that state that "The Design Temperature shall be the temperature field to which the Composite Core Component is exposed that, in combination with the Design Fast Flux, results in the highest use of the Composite Core Component."

y. HHB-3141, Chemical Attack and Oxidation

(1) The NRC staff is not endorsing the weight change limit of 1% in HHB-3141(a) which distinguishes between oxidized and not oxidized composite components. The Designer should determine their own weight change at which a component is considered oxidized and justify that limit.

(2) The NRC staff endorses HHB-3141(b), Strength Reduction, with the following exception: Designers should consider secondary and irradiation loads without removing the low strength region when it results in the highest use of the Composite Core Component.

(3) The NRC staff is not endorsing the provisions of HHB-3141(c), Geometry Reduction, that sets the geometry reduction weight loss limit of 30% for both chemical attack and strength calculations. Designers should determine the amount of weight loss above which the region should be regarded as completely removed from the structure and justify that the limit is adequate for the design-specific chemical attack and strength calculations.

(4) HHB-3141(d), Alternative Methodology, states that chemical attack to high weight loss (>1%) occurring simultaneously with significant irradiation (>0.25 dpa) is not permitted. Simultaneous chemical attack to weight loss >1% and irradiation to >0.25 dpa is outside the scope of the code requirements and Designers should develop and use an alternative methodology to assess a Composite Core Component subject to these conditions.

z. HHB-3142.1, Irradiated Composite Core Components Classification

(1) The NRC staff is not endorsing the provisions of HHB-3142.1 that classify Composite Core Components as nonirradiated when the cumulative fast neutron irradiation fluence is (a) less than 0.25 dpa for C-C composites and (b) less than 1.0 dpa for SiC-SiC composites. The Designer shall develop and justify the minimum dose limit for classifying composites as nonirradiated.

aa. HHB-3143, Abrasion and Erosion

(1) The NRC staff is not endorsing the provisions of HHB-3143(b) that set the mean gas flow velocity limit of 100 meters per second (330 feet per second) for evaluating the effects of erosion on the Composite Core Component design. Designers should determine the mean gas flow velocity limit above which an evaluation of erosion is necessary and justify that the

limit is adequate for the Composite Core Component design. Designers should also consider erosion from all sources.

bb. HHB-3213, Basis for Determining Stresses

(1) This paragraph is describing the assessment methodology of simplified assessment. The NRC places conditions on the simplified assessment as seen in ‘ee’ below.

cc. HHB-3214.1, Loading Mode Stress

(1) This subparagraph is part of simplified assessment methodology. The NRC conditions use of the simplified assessment as seen in ‘ee’ below.

dd. HHB-3214.2, Maximum Loading Mode Stress

(1) This subparagraph is part of simplified assessment methodology. The NRC conditions use of the simplified assessment as seen in ‘ee’ below.

ee. HHB-3220, Stress Limits for Composite Core Components

(1) The simplified assessment may only be used with appropriate justification. Alternatively, applicants and licensees may develop their own analysis methods and validated failure criteria for those failure modes identified by the Designer.

ff. HHB-III-3100, As-Manufactured Ceramic Composite Material

(1) The NRC staff is not endorsing the provision of HHB-III-3100 that states that the temperature dependence of only one strength parameter shall be determined for each anisotropic direction, and the other strength parameters may be assumed to change by the same relative fraction. The temperature dependence of each strength parameter should be measured, or the proportionality of strength parameters should be justified.

gg. Nonmandatory Appendix HHB-B, Composition, Structure, Manufacture, and Properties of Ceramic Matrix Composites

(1) The NRC staff is not endorsing Nonmandatory Appendix HHB-B because it is for information only and does not provide guidance for applicants or licensees.

hh. Nonmandatory Appendix HHB-C, Fracture and Damage Mechanisms in SiC-SiC CMCs

(1) The NRC staff is not endorsing Nonmandatory Appendix HHB-C because it is for information only and does not provide guidance for applicants or licensees.

ii. Nonmandatory Appendix HHB-D, Carbon-Carbon (C-C) Composite Materials

(1) The NRC staff is not endorsing Nonmandatory Appendix HHB-D because it is for information only and does not provide guidance for applicants or licensees.

jj. Nonmandatory Appendix HHB-E, Carbon-Carbon (C-C) Composite Materials Irradiation and Environmental Effects

(1) The NRC staff is not endorsing Nonmandatory Appendix HHB-E because it does not provide guidance for applicants or licensees.

2. ASME Code, Section III, Division 5, Code Cases

The NRC staff endorses the Code Cases listed in Table 4 below, without limitations, for application in the design and construction of high-temperature reactors.

Table 4. Acceptable ASME Code, Section III, Division 5, Code Cases

CODE CASE NUMBER	CODE CASE TITLE
N-812-1	Alternate Creep-Fatigue Damage Envelope for 9Cr-1Mo-V Steel
N-861-2	Satisfaction of Strain Limits for Division 5 Class A Components at Elevated Temperature Service Using Elastic-Perfectly Plastic Analysis
N-862-2	Calculation of Creep-Fatigue for Division 5 Class A Components at Elevated Temperature Service Using Elastic-Perfectly Plastic Analysis
N-872	Use of 52Ni-22Cr-13Co-9Mo Alloy 617 (UNS N06617) for Low Temperature Service Construction, Section III,Division 5.”
N-924	Design Rules and Limits for Load-Controlled Stresses for Class A Components at Elevated Temperature Service Using Elastic-Perfectly Plastic and Simplified Inelastic Analyses

The NRC staff endorses the Code Cases listed in Table 5 below, with limitations, for application in the design and construction of high-temperature reactors, except where ASME identifies portions of the Code Case as being in the course of preparation. The NRC staff is unable to review those sections identified as in the course of preparation to determine whether they are acceptable, and therefore the staff does not endorse them. The staff bases for the limitations in Table 5 are presented in Section B “Discussion” under subheading “Bases for NRC Staff Regulatory Guidance Positions”.

Table 5. Conditionally Acceptable ASME Code, Section III, Division 5, Code Cases

CODE CASE NUMBER	CODE CASE TITLE/LIMITATION NUMBER/LIMITATION
N-898-1	*Use of Alloy 617 (UNS N06617) for Class A Elevated Temperature Service Construction Section III, Division 5*
	1-CC-N-898-1- HBB-Z-1212.3, Accumulated Damage – The NRC staff is not endorsing this paragraph
	2-CC-N-898-1- HBB-Z-1326, Alloy 617 – The NRC staff is not endorsing this paragraph

N-940[6]	*Alternate Rules for Nondestructive Examination and Testing of Items Commensurate with their Contribution to Safety or Risk* *Section III, Division 5*
	<u>1-CC-N-940</u> The ultrasonic examination should be performed using encoded ultrasonic examination technology that produces an electronic record of the ultrasonic responses indexed to the probe position, permitting off-line analysis of images built from the combined data. The data from the encoded scans shall be preserved for the life of the plant. Where physical obstructions prevent the use of encoded ultrasonic examination technology, non-encoded ultrasonic examination technology may be used. The basis for the non-encoded examination shall be documented.
	<u>2-CC-N-940</u> The ultrasonic examination should be qualified using ASME Section V, Article 14 Intermediate Rigor on test specimens that are of the same materials and similar size and thickness for the welds being examined.
	<u>3-CC-N-940</u> Progressive sampling under Nonmandatory Appendix B should not be applied to elevated temperature service (temperature > 750°F).
	<u>4-CC-N-940</u>For applications of progressive sampling under Nonmandatory Appendix B, the initial sample size should be one of the following: (1) a population justified statistically to provide 95% confidence that 5% or fewer of the welds contain defects, or (2) a lesser initial sample justified as an alternative approach as described in footnote 6 below, or (3) for instances where a designer does not prefer to use the statistical justification in (1) or develop an alternative approach in (2), 50% random sampling is acceptable.
	<u>5-CC-N-940</u> The definition of a moderate energy piping system for fluid systems with a service fluid other than water under -7000(b)(2) of this Code Case is subject to NRC review and approval. The justification for a definition of moderate energy piping for a technology should consider the potential impacts on other SSCs from failure of the piping system based on the operating conditions and characteristics of the fluid. Similar considerations for light-water reactors can be found in Branch Technical Position (BTP) 3-3, Revision 3 (ML070800027).

[6] Alternative approaches to those specified in conditions 1-CC-N-940 through 6-CC-N-940 may be proposed with justification subject to NRC review and approval. Justification should be based on the ability to meet the reliability and capability targets for the SSC. The justification can also include consideration of performance monitoring approaches, such as surveillance methods, in-service inspection, and continuous monitoring. Continuous monitoring of an SSC's ability to perform its function, such as online monitoring for cracking, pressure, temperature, or chemistry changes to indicate a leak or loss of boundary integrity, could bolster such a justification.

	6-CC-N-940 Nonmandatory Appendix D should be implemented consistent with paragraph 137.7.1 from ASME BPVC B31.1: 137.7.1 When specified by the owner, an initial service test and examination is acceptable *when other types of tests are not practical or when leak tightness is demonstrable due to the nature of the service*. One example is piping where shutoff valves are not available for isolating a line and where temporary closures are impractical. Others may be systems where during the course of checking out of pumps, compressors, or other equipment, ample opportunity is afforded for examination for leakage prior to full scale operation. In particular, an initial service leak test shall be used only "when other types of tests are not practical or when leak tightness is demonstrable due to the nature of the service."

D. IMPLEMENTATION

The NRC staff may use this RG 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 RG 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,” (Ref. 11), nor does the NRC staff intend to use the guidance to affect the issue finality of an approval under 10 CFR Part 52, “Licenses, Certifications, and Approvals for Nuclear Power Plants.” 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 RG in a manner inconsistent with the discussion in this Implementation section, then the licensee may file a backfitting or forward fitting appeal with the NRC in accordance with the process in Management Directive 8.4.

REFERENCES [7]

1. American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) Code, Section III, "Rules for Construction of Nuclear Facility Components," Division 5, "High Temperature Reactors," 2023 Edition, New York, NY.[8]

2. *U.S. Code of Federal Regulations* (CFR), "Domestic Licensing of Production and Utilization Facilities" Part 50, Chapter 1, Title 10, "Energy"

3. 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants"

4. U.S. Nuclear Regulatory Commission (NRC), NUREG-2245, "Technical Review of the 2017 Edition of ASME Section III, Division 5, "High Temperature Reactors"," Washington, DC. (ADAMS Accession No ML22101A208)

5. NRC, Regulatory Guide (RG) 1.232, Revision 0, "Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors," Washington, DC, April 2018. (ML17325A611)

6. NRC, RG 1.26, Revision 5, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," Washington, DC, February 2017. (ML16286A590)

7. NRC, Interim Staff Guidance, "Material Compatibility for non-Light Water Reactors," Washington, DC, February 2024. (ML23188A178)

8. ASME BPV Code, Section III, Division 1, "Rules for Construction of Nuclear Facility Components," 2023 Edition, New York, NY

9. NRC, RG 1.87, Revision 2, "Acceptability of ASME Code, Section III, Division 5, "High Temperature Reactors"," Washington DC, January 2023. (ML22101A254)

10. NRC, Technical Letter Report RES/DE/REB-2022-01 "Review of Code Cases Permitting Use of Nickel-Based Alloy 617 in Conjunction with ASME Section III, Division 5," Washington, DC, January 2022. (ML22031A137)

11. NRC, Management Directive (MD) 8.4, "Management of Backfitting, Forward Fitting, Issue Finality, and Information Requests," Washington, DC. (ML18093B087)

[7] Publicly available NRC published documents are available electronically through the NRC Library on the NRC's public website at http://www.nrc.gov/reading-rm/doc-collections/ and through the NRC's Agencywide Documents Access and Management System (ADAMS) at http://www.nrc.gov/reading-rm/adams.html. For problems with ADAMS, contact the Public Document Room staff at 301-415-4737 or (800) 397-4209, or email pdr.resource@nrc.gov. The NRC Public Document Room (PDR), where you may also examine and order copies of publicly available documents, is open by appointment. To make an appointment to visit the PDR, please send an email to PDR.Resource@nrc.gov or call 1-800-397-4209 or 301-415-4737, between 8 a.m. and 4 p.m. eastern time (ET), Monday through Friday, except Federal holidays.

[8] Copies of American Society of Mechanical Engineers (ASME) standards may be purchased from ASME, Two Park Avenue, New York, NY 10016-5990; telephone (800) 843-2763. Purchase information is available through the ASME Web-based store at https://www.asme.org/publications-submissions/publishing-information.

12. L. Y. Wang, R. Y. Luo, G. yuan Cui, and Z. feng Chen, “Oxidation resistance of SiCf/SiC composites with a PyC/SiC multilayer interface at 500 °C to 1100°C,” Corros. Sci., vol. 167, no. November 2019, p. 108522, 2020, doi: 10.1016/j.corsci.2020.108522

13. NRC, “Nuclear Regulatory Commission International Policy Statement,” *Federal Register*, Vol. 79, No. 132, July 10, 2014, pp. 39415–39418.

14. NRC, MD 6.6, “Regulatory Guides,” Washington, DC, May 2, 2016. (ML18073A170)

15. ASME B31.1, “Power Piping” 2022 Edition, New York, NY

16. ASME BPV Code, Section XI, Division 1, Code Case N-831-1, “Ultrasonic Examination in Lieu of Radiography for Welds in Ferritic or Austenitic Pipe” 2021 Edition, New York, NY

17. ASME BPV Code, Section V, “Nondestructive Examination” 2021 Edition, New York, NY

18. ASME B31.3, “Process Piping” 2022 Edition, New York, NY

19. International Atomic Energy Agency, “Safety of Nuclear Power Plants: Design, Specific Safety Requirements,” IAEA Safety Standards Series No. SSR-2/1 (Rev. 1), Vienna, Austria, 2016.[9]

20. NRC, RG 1.233, “Guidance for a Technology-Inclusive Risk-Informed, and Performance-Based methodology to Inform the Licensing Basis and Content of Applications for Licensees, Certifications, and approvals for Non-Light Water Reactors,” Washington, DC, June 2020. (ML20091L698)

[9] Copies of International Atomic Energy Agency (IAEA) documents may be obtained through its Web site: www.IAEA.Org/ or by writing the International Atomic Energy Agency, P.O. Box 100 Wagramer Strasse 5, A-1400 Vienna, Austria.

APPENDIX A

HIGH TEMPERATURE REACTOR QUALITY GROUP CLASSIFICATION

A-1. Introduction

The nuclear industry has several means for the safety classification of components available. These include (1) the traditional means outlined in the Title 10 of the *Code of Federal Regulations* (10 CFR) using the definition of *safety-related structures, systems, and components (SSCs)* in 10 CFR 50.2 (Ref. A-1), (2) the risk-informed classification system in 10 CFR 50.69, "Risk-informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors," and (3) the method in Nuclear Energy Institute (NEI) 18-04, Revision 1, "Risk-Informed Performance-Based Technology-Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development," (Ref. A-2), endorsed in 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," (Ref. A-3). The guidance in this appendix establishes quality group assignments of mechanical systems and components of non-LWRs acceptable to the U.S. NRC staff for all the safety classification methods mentioned above and is intended to provide guidance on selecting an appropriate design standard once the classification methods are used to determine the classification of each system and component. Quality groups are a quality classification system to provide applicants and licensees with guidance for satisfying design criteria and assigning specific quality standards.

In establishing standards acceptable to the NRC staff, it is not possible to know all the design details associated with future designs. There may be some instances where the standards established in this appendix may be overly conservative or possibly require supplementation for a specific design. As such, the NRC staff will evaluate an applicant's implementation of this appendix on a case-by-case basis to determine if the proposals are appropriate for the specific design.

ASME Boiler and Pressure Vessel (BPV) Code (ASME Code), Section III, Division 5, "High temperature Reactors," (Ref. A-4) is endorsed in the main body of this RG. The additional standards referenced in this Appendix are likely to be appropriate for the identified use. The staff will evaluate the application of these standards for a particular design and related justification on a case-by-case basis.

A-2. Safety Classification Categories

Traditional Approach

In the traditional approach consistent with the current NRC regulations, SSCs are classified as either SR or non-safety-related (NSR). Those SSCs which maintain the integrity of the reactor coolant pressure boundary, are relied upon to shut down the reactor and maintain it in a safe shut down condition, or prevent or mitigate the consequences of an accident that could result in potential offsite exposures comparable to the applicable guideline exposures in 10 CFR 50.34(a)(1) are designated as SR. All other components are designated as NSR.[1] Under the traditional approach for LWRs, as described in RG 1.26 "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," (Ref. A-5), once mechanical SSCs are determined to be SR or NSR, they are further categorized into Quality Groups, either A, B, C, or D. Quality groups A, B, and C are SR, and Quality Group D is NSR. The quality groups are aligned with the ASME Code classification system in Section III, Division 1. Quality Group A is aligned with ASME Class 1 for reactor coolant system pressure boundary components; Quality Group B is aligned with ASME Class 2, for systems that provide engineered safety features or emergency core cooling functions; and Quality Group C is for the remaining SR systems that do not meet the criteria for assignment in Quality Groups A or B. These Quality Group C components are generally the components making up the support systems and ultimate heat sink for the reactor, including component cooling and service water systems. Quality Group C is aligned with ASME Class 3. RG 1.26 also assigns Quality Group C to SSCs, other than SSCs in radioactive waste systems, whose postulated failure would result in conservatively calculated potential offsite doses exceeding the regulatory limit of 0.1 rem total effective dose equivalent (TEDE) set by 10 CFR Part 20 "Domestic Licensing of Production and Utilization Facilities," (Ref. A-6).

SSCs that are NSR and not otherwise provided a Quality Group classification may have some special function, such as providing defense-in-depth or containing radioactive material. RG 1.26 assigns Quality Group D to water- and steam- containing components that are not part of the reactor coolant pressure boundary or included in Quality Groups B or C but are part of systems that contain or may contain radioactivity. RG 1.143 "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants" (Ref. A-7), [2] provides information related to the classification of SSCs in radioactive waste management systems that fall within the scope of that RG. While such SSCs do not meet the criteria for SR SSCs, there is still a need to ensure component integrity. These RGs endorse the following standards, among other special treatments, as acceptable to assure the integrity of SSCs performing the NSR functions within their scope: ASME Code, Section VIII, Division 1, "Rules for Construction of Pressure Vessels," (Ref. A-8); ASME Code, Section VIII, Division 2, "Alternative Rules," (Ref. A-9); ASME B31.1, "Power Piping" (Ref. A-10); and ASME B31.3, "Process Piping," (Ref. A-11). These standards include high-temperature operating conditions within their scope that may be appropriate for non-LWRs; the adequacy of these standards may be addressed during the review of an application for a specific design.

This appendix addresses pressure-retaining components, core components, and supports of high-temperature reactors. The guidance in RG 1.26 should be used for pressure-retaining components containing water, steam, or radioactive material in light-water-cooled nuclear power plants. Other systems not covered by this RG 1.87, such as instrument and service air; diesel engines, their generators, and auxiliary support systems; diesel fuel; emergency and normal ventilation; fuel handling; and radioactive waste management systems, should be designed, fabricated, erected, and tested to quality standards commensurate with the safety function to be performed.

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1 Current NRC regulations define "safety-related SSCs" in § 50.2 and "important to safety SSCs" in Appendix A to Part 50. Important to safety SSCs perform the functions required by the General Design Criteria in Part 50, Appendix A, or other substantive regulations, and may or may not be safety related. Section 50.69 uses the term "nonsafety-related SSCs" to define categories of SSCs based on risk (categories "RISC-2" and "RISC-4"). Under Part 50, non-safety-related SSCs include important to safety SSCs and also include SSCs that do not perform any safety function required by NRC regulations or credited in the safety analysis

2 RG 1.143 provides guidance on the design and quality classification of solid, liquid, and gaseous radwaste system and steam generator blowdown SSCs. RG 1.143 uses 500 millirem (0.5 rem) as a dose criterion for classification. This criterion was based on the 10 CFR Part 20 dose limit before 1994, when the NRC revised it down to 100 millirem (0.1 rem). While the NRC did not update RG 1.143 to reflect the current requirement, the staff did update NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," (Ref. A-12) to the 100 millirem criterion.

In the transition to the establishment of quality groups for non-LWRs, design differences affect the traditional safety classification process and the design rules applied to SR SSCs. The NRC staff recognizes that the definition of *safety-related structure, system, or component* in 10 CFR 50.2 may not be fully applicable to the design of all high-temperature reactors because the design may not include components that satisfy the definition of *reactor coolant pressure boundary* in 10 CFR 50.2. An applicant for such a design may need to obtain exemptions from the definition of *safety-related structure, system, or component* to use the traditional safety classification process.

The design rules for high-temperature reactor mechanical components provided in ASME Code, Section III, Division 5, use only two classes, Class A and Class B, rather than the three classes identified in Division 1. Class A rules are the more rigorous rules for the design of elevated temperature applications and should be used when it is important that the component not suffer a failure. Class B is less rigorous in design and should be used when the consequences of a SR component failure are less significant. The differences in the classes involve how creep and thermal cycling are treated. For SR pressure-retaining components and supports, the ASME Code, Section III should be used. Both Division 1, "Rules for Construction of Nuclear Facility Components" (Ref. A-13) and Division 5 have rules appropriate for ferritic and austenitic design temperatures below 370 °C (700 °F) and 425 °C (800 °F), respectively, and Division 5 should be used for ferritic and austenitic design temperatures above 370 °C (700 °F) and 425 °C (800 °F), respectively. For high-temperature reactor SR components that operate below these temperatures, Division 5, which refers to the rules in Division 1, should be followed. The application of standards other than ASME Code, Section III may be justified on a case-by-case basis.

This RG does not endorse specific standards for NSR components of high-temperature reactors. ASME Code, Section VIII, Division 1 or 2 for vessels and ASME B31.1 or ASME B31.3 for power piping and process piping, respectively, are likely appropriate for the design of NSR mechanical components within the scope of these standards that need special treatment, such as for systems providing defense-in-depth or containing radioactive material. Application of standards for NSR SSCs with special treatment may be justified on a case-by-case basis. SSCs that are NSR and do not meet the criteria for special treatment are left to the applicant to specify any standards for design and fabrication.

Risk-Informed Approach

The NRC has developed a voluntary classification method in 10 CFR 50.69 that establishes four categories for SSCs, and the NRC staff has determined that these categories can be used for SSC safety classification for non-LWR designs.[3] The categories listed below are based on the traditional approach discussed above, but then take into consideration the safety significance of the functions performed using the guidance in RG 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance" (Ref. A-14):

- SR SSCs that perform safety-significant functions (RISC-1),
- NSR SSCs that perform safety-significant functions (RISC-2),
- SR SSCs that perform low safety-significant functions (RISC-3), and
- NSR SSCs that perform low safety-significant functions (RISC-4)

SR SSCs that perform safety significant functions (RISC-1) should be designed to standards in accordance with Advanced Reactor Design Criterion (ARDC)-1 of RG 1.232 and 10 CFR Part 50,

[3] Section 50.69 uses these classifications to determine which SSCs the NRC can approve for alternative, risk-informed treatment. This RG does not provide guidance on such alternative treatment under 10 CFR 50.69, which requires NRC approval.

Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants” (Ref. A-15). The NRC staff has determined that ASME Code, Section III, Division 5, provides acceptable standards for SR SSCs that perform safety-significant functions (RISC-1) in high-temperature environments, with the distinction between Class A and Class B criteria determined by the safety significance of the component.

NSR SSCs are not typically subject to the nuclear quality standards associated with a quality assurance program that complies with Part 50, Appendix B. However, for NSR SSCs that perform safety significant functions (RISC-2), some type of augmented quality is warranted, and a design standard should be used that ensures a high degree of reliability of the SSC, consistent with ARDC-1 of RG 1.232. ASME Code, Section VIII and ASME B31.1 or B31.3, which RG 1.26 and RG 1.143 endorse for similar purposes, are likely to be appropriate standards to apply to NSR SSCs in high-temperature environments that need special treatment. The application of standards other than ASME Code, Section III, may be justified on a case-by-case basis.

SR SSCs that perform low safety-significant functions (RISC-3) may have alternative requirements established under 10 CFR 50.69. If RISC-3 components are identified, ASME Section VIII, B31.1, or B31.3 standards encompassing mechanical component design for high-temperature applications may be justified on a case-by-case basis.

The assignment of appropriate design standards for NSR SSCs that perform low safety-significant functions (RISC-4) is left to the designers and owners because these SSCs are generally used to support commercial aspects of the facility.

License Modernization Project (LMP) Approach (RG 1.233)

The NRC staff issued guidance for using a technology-inclusive, risk-informed, and performance-based methodology to inform the content of applications for licenses, certifications, and approvals for non-LWRs in RG 1.233. RG 1.233 endorses, with clarifications, the principles and methodology in NEI 18-04, Revision 1, as one acceptable method for safety classification of SSCs for non-LWRs.

The LMP approach under RG 1.233 focuses on the functional performance of SSCs, which may in some cases lead to specific SSCs possessing both SR and NSRST functions (e.g. SR for one function and NSRST for a different function). In these cases, the overall classification for an SSC is determined by its function with the highest safety classification. Accordingly, the staff guidance in Table A-1 is based on the overall classification for an SSC. However, an SSC with functions of varying safety classification may be able to justify a different code and standard than the application of Table A-1 to the overall classification based on the specific details of the plant design and functions of that SSC within the plant.

NEI 18-04 includes a methodology to classify SSCs as either SR, NSR with special treatment, or NSR with no special treatment. NEI 18-04[4] gives the following definitions for these terms:

- Safety-Related (SR)
  - SSCs selected by the designer from the SSCs that are available to perform the required safety functions to mitigate the consequences of design-basis events to within the licensing basis event

[4] The methodology in NEI 18-04 includes a definition and means to identify SR SSCs for non-LWRs different from that used in the deterministic approaches for LWRs. NEI 18-04 includes a glossary to help alleviate some of the issues that will arise because of differences in terminology. Applicants referencing RG 1.233 should use the terminology in NEI 18-04 and, as needed, identify exceptions to and exemptions needed from NRC regulations.

frequency-consequence (F-C) target (described in NEI 18-04), and to mitigate design-basis accidents that only rely on the SR SSCs to meet the dose limits of Title 10 of the Code of Federal Regulations (10 CFR) 50.34, "Contents of Applications; Technical Information," using conservative assumptions

  - SSCs selected by the designer and relied on to perform required safety functions to prevent the frequency of beyond-design-basis events with consequences greater than the 10 CFR 50.34 dose limits from increasing into the design-basis event region and beyond the F-C target

- Non-safety-Related with Special Treatment (NSRST)

  - NSR SSCs relied on to perform risk-significant functions; risk-significant SSCs are those that perform functions that prevent or mitigate any licensing basis event from exceeding the F-C target or make significant contributions to the cumulative risk metrics selected for evaluating the total risk from all analyzed licensing basis events

  - NSR SSCs relied on to perform functions requiring special treatment for defense-in-depth adequacy

- Non-safety-Related with No Special Treatment (NST)

  - all other SSCs (with no special treatment required)

SR SSCs should be designed to nuclear codes and standards. ASME Code, Section III, Division 5 is acceptable to the NRC, with the conditions noted in this RG. Class A rules are the more rigorous rules for the design of components that operate at elevated temperature conditions and should be used when a component performs safety-significant functions. Class B is less rigorous in design and should be used only when the component performs less safety-significant functions.

NSRST SSCs should also be designed to appropriate standards. ASME Section VIII, B31.1, or B31.3 are standards encompassing mechanical component design for high-temperature applications that may be justified on a case-by-case basis.

The assignment of appropriate design standards for NSR components with no special treatment are left to the designers and owners.

A-3. Quality Standards

Advanced Reactor Design Criterion 1 in RG 1.232, "Guidance for Developing Principal Design Criteria for Non-Light-Water Reactor," (Ref. A-16), states that SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. ARDC 1 also states that where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. The NRC staff finds ASME Code, Section III, Division 5, acceptable for ARDC 1 with conditions as documented in this RG for use in high-temperature applications in nuclear reactor designs. The ASME Code contains two design classes for metallic components, Class A and Class B. The provisions for creep and cyclic loading are treated differently between the two classes, with Class A being the more stringent of the two. The ASME Code, Section III, Division 5 rules for Class A rely heavily on the rules for ASME Code, Section III, Division 1, Class 1, and apply additional rules for addressing creep and thermal transients. The ASME Code, Section III, Division 5, Class B rules rely on ASME Code, Section III, Division 1, Class 2. Class B rules are not as rigorous as Class A and do not include thermal transient rules and should only be used when the consequences of a failure are less significant. ASME Code, Section VIII, Division 1 or 2, along with piping codes ASME B31.1 or ASME B31.3 are likely appropriate standards to use for NSR SSCs within their scope that warrant special treatment for functions such as defense-in-depth or to maintain exposure to the public from failures in radionuclide containing systems within regulatory limits. Application of standards for special treatment of NSR SSCs may be justified on a case-by-case basis. For NSR SSCs that require no special treatment, the selection of standards is left to the designers and owners.

A-4. Quality Group Classifications

For high-temperature applications of non-LWRs, this section describes an acceptable method to map SSC safety classifications to appropriate quality standards. To accomplish that goal, the staff defined three quality groups based on the classification of the SSCs determined using any of the methods described above. The quality groups are determined by the classification of the SSC as either SR or NSR, and the safety significance of the SSC functions. SSCs classified as RISC-1 using the 10 CFR 50.69 classification process or SR by the traditional or LMP approaches are divided into two quality group classifications. Under the traditional approach, this division is based on the significance of the SSC function, and under § 50.69 or the LMP this division is based on the degree of safety-significance. The most safety-significant SSCs in these classifications should be assigned to Quality Group A. The SSCs within these classifications with less safety significance, yet still considered safety-significant in the risk-informed 10 CFR 50.69 and LMP classification processes, may be assigned to Quality Group B. SSCs classified as NSR but perform an important to safety function under the traditional approach or are considered safety-significant in the risk-informed 10 CFR 50.69 and LMP classification processes are assigned to Quality Group C. The SSCs classified as SR with low safety significance (RISC-3) using the 10 CFR 50.69 classification process may also be assigned to Quality Group C.[5] The SSCs classified as NSR without an important to safety function are not assigned to a quality group because the owner or designer establishes the quality standards. Assignment of appropriate design standards to the SR classification will depend upon the consequence of component failure and the level of quality assurance necessary. The consequences of component failure will indicate the appropriate ASME Code design class for the SSC. Core support structures should be classified as SR with the highest safety significance because these components ensure the core configuration is maintained in an analyzed configuration.

Table A-1, "Classification and Standards Applicable to Components in High Temperature Reactors," may be used to identify an appropriate standard for the design and fabrication of safety-significant high-temperature reactor components. The traditional approach is based on evaluation of SSC functions considering the definition of SR SSCs and the categories of functions in §§ 50.55a(c)-(e). The categorization process outlined in 10 CFR 50.69(c) uses the definition of SR at the system or structure level and consider (1) the results of a design-specific probabilistic risk analysis, (2) an evaluation of functional significance, and (3) maintenance of defense-in-depth. The categorization process under the LMP approach employs similar considerations to complete SSC classification as SR or NSRST (or NST). Each classification process is subject to NRC review.

After the selected classification process has been completed, the SSCs should be subdivided into one of the three Quality Groups as described above. Quality Groups A and B align with the ASME Code Classes A and B, respectively, of Section III, Division 5, of the ASME Code endorsed by this RG. The

[5] Note that the LMP risk-informed classification process does not have a comparable category because only SSCs with safety significant functions are considered SR. Additionally, all SSCs classified as SR using the traditional classification method are considered Quality Group A or B because the classification process does not include risk-informed elements to fully consider defense in depth.

standards identified in Table A-1 for Quality Group C, RISC 2, RISC-3, or NSRST components (i.e., ASME Section VIII, and B31.1 or B31.3) are standards encompassing mechanical component design for high-temperature applications that may be justified on case-by-case basis for components classified in those groups. Table A-1 represents the design standards that the NRC has determined are appropriate for the different categorization methods described in this appendix without having specific design information available for a reactor design. This does not mean that other codes or standards are not acceptable, but the NRC has not generically evaluated other codes or standards at this time. There may be instances where deviations from the recommendations in Table A-1 can be justified based on the specifics of the design.

Table A-1. Classification and Standards Applicable to Components in High Temperature Reactors

<table>
<tr><th>Classification Method</th><th colspan="3">Component Classification</th></tr>
<tr><td>Traditional</td><td>Quality Group A</td><td>Quality Group B</td><td>Quality Group C</td></tr>
<tr><td>Risk-Informed (10 CFR 50.69)[6]</td><td>RISC-1</td><td>RISC-1</td><td>RISC-2, RISC-3</td></tr>
<tr><td>Licensing Modernization Project (LMP) Approach (RG 1.233)</td><td>SR</td><td>SR</td><td>NSRST</td></tr>
<tr><td></td><td colspan="2">SR Quality Design Standards</td><td>Important to Safety Design Standards</td></tr>
<tr><td>Components</td><td colspan="2"></td><td></td></tr>
<tr><td>Pressure Vessels</td><td rowspan="6">ASME Code, Section III, Division 5, Class A</td><td rowspan="6">ASME Code, Section III, Division 5, Class B</td><td>ASME Code, Section III, Division 5 or Industrial Codes with appropriate justification[7]</td></tr>
<tr><td>Piping</td><td>ASME Code, Section III, Division 5 or Industrial Codes with appropriate justification[8]</td></tr>
<tr><td>Pumps</td><td></td></tr>
<tr><td>Valves</td><td>ASME Code, Section III, Division 5 or Industrial Codes with appropriate justification[8]</td></tr>
<tr><td>Atmospheric Storage Tanks</td><td></td></tr>
<tr><td>Storage Tanks (0-15 pounds per square inch gauge)</td><td>ASME Code, Section III, Division 5 or Industrial Codes with appropriate justification[7]</td></tr>
<tr><td>Metallic Core Support Structures</td><td>ASME Code, Section III, Division 5, Subsection HG</td><td>N/A</td><td></td></tr>
<tr><td>Nonmetallic Core Components</td><td>ASME Code, Section III, Division 5, Subsection HH</td><td>N/A</td><td></td></tr>
</table>

6 Alternative treatment under 10 CFR 50.69 for SSCs categorized as RISC-1, RISC-2, RISC-3, or RISC-4 requires NRC review and approval in accordance with 10 CFR 50.69.

7 These standards may include ASME Code, Section III, Division 5, which is endorsed by NRC. Codes that have not been endorsed by NRC, such as ASME Code, Section VIII, Division 1 and Division 2, or other alternate standards, may be used with appropriate justification. The applicant should justify how codes that have not been endorsed by the NRC as well as any special treatments are appropriate for the SSC.

8 These standards may include ASME Code, Section III, Division 5, which is endorsed by NRC. Codes that have not been endorsed by NRC, such as, ASME B31.1/B31.3 or other alternate standards, may be used with appropriate justification. The applicant should justify how codes that have not been endorsed by the NRC as well as any special treatments are appropriate for the SSC.

APPENDIX A REFERENCES[9]

A-1. *U.S. Code of Federal Regulations* (CFR), "Domestic Licensing of Production and Utilization Facilities," Part 50, Chapter 1, Title 10, "Energy."

A-2. Nuclear Energy Institute (NEI), Technical Report 18-04, "Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development," Revision 1, Washington, DC, August 2019. (ADAMS Accession No, ML19241A472).[10]

A-3. U.S. Nuclear Regulatory Commission (NRC), Regulatory Guide (RG) 1.233, Revision 0 "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," Washington, DC, June 2020. (ML20091L698).

A-4. ASME, Boiler and Pressure Vessel Code, Section III, Division 5, "High Temperature Reactors," 2017 Edition, New York, NY.

A-5. NRC, RG 1.26, Revision 5, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," Washington, DC, February 2017. (ML16286A590).

A-6. 10 CFR Part 20, "Domestic Licensing of Production and Utilization Facilities."

A-7. NRC, RG 1.143, Revision 2, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants," Washington DC, November 2001. (ML013100305).

A-8. ASME, Boiler and Pressure Vessel Code, Section VIII, Division 1, "Rules for Construction of Pressure Vessels," New York, NY.

A-9. ASME, Boiler and Pressure Vessel Code, Section VIII, Division 2, "Alternative Rules," New York, NY

A-10. ASME Standard B31.1, "Power Piping," New York, NY.

A-11. ASME Standard B31.3, "Process Piping," New York, NY.

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9 Publicly available NRC published documents are available electronically through the NRC Library on the NRC's public website at http://www.nrc.gov/reading-rm/doc-collections/ and through the NRC's Agencywide Documents Access and Management System (ADAMS) at http://www.nrc.gov/reading-rm/adams.html. For problems with ADAMS, contact the Public Document Room staff at 301-415-4737 or (800) 397-4209, or email pdr.resource@nrc.gov. The NRC Public Document Room (PDR), where you may also examine and order copies of publicly available documents, is open by appointment. To make an appointment to visit the PDR, please send an email to PDR.Resource@nrc.gov or call 1-800-397-4209 or 301-415-4737, between 8 a.m. and 4 p.m. eastern time (ET), Monday through Friday, except Federal holidays.

10 Publications from the Nuclear Energy Institute (NEI) are available at its Web site: http://www.nei.org/ or by contacting the headquarters at Nuclear Energy Institute, 1776 I Street NW, Washington DC 20006-3708, Phone: 202-739-800, Fax 202-785-4019.

A-12. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Washington, DC.

A-13. American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code, Section III, Division 1, "Rules for Construction of Nuclear Facility Components," 2017 Edition, New York, NY.[11]

A-14. NRC, RG 1.201, Revision 1, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance," Washington, DC, May 2006. (ML061090627).

A-15. 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants"

A-16. NRC, RG 1.232, Revision 0, "Guidance for Developing Principal Design Criteria for Non-Light-Water Reactor," Washington, DC, April 2018. (ML17325A611).

[11] Copies of American Society of Mechanical Engineers (ASME) standards may be purchased from ASME, Two Park Avenue, New York, NY 10016-5990; telephone (800) 843-2763. Purchase information is available through the ASME Web-based store at https://www.asme.org/publications-submissions/publishing-information.