U.S. NUCLEAR REGULATORY COMMISSION

REGULATORY GUIDE 1.87, REVISION 2

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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.
    - 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.

Written suggestions regarding this guide or development of new guides may be submitted through the NRC's public Web site 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. 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 Web site 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 http://www.nrc.gov/reading-rm/adams.html, under ADAMS Accession Number (No.) ML22101A263. The regulatory analysis may be found in ADAMS under Accession No. ML21091A277. The associated draft guide DG-1380, may be found in ADAMS under Accession No. ML21091A276, and the staff responses to the public comments on DG-1380 may be found under ADAMS Accession No. ML22101A267.

- 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 and N-862.

- 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.

- 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 (PDC) to the effect, in part, that SSCs important to

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

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.

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 supported by a basis for the issuance or continuance of a permit or license by the Commission.

Paperwork Reduction Act

This RG provides voluntary guidance for implementing the mandatory information collections in 10 CFR Parts 50 and 52 that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.). These information collections were approved by the Office of Management and Budget (OMB), under control numbers 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 (3150-0011 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 Issuance

This revision (Revision 2) updates the guidance to endorse, with exceptions and limitations, the 2017 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 of high temperature reactors. The NRC staff is also endorsing use of certain values in the 2019 Edition of ASME Code, Section II, "Materials," Part D, "Properties (Metric)" (Ref. 7) and Mandatory Appendix HBB-I-14 of the 2019 Edition of the ASME Code, Section III, Division 5 for limited use. This revision of the guide also endorses the Code Cases N-861, N-862, N-872, and N-898 related to ASME Code, Section III, Division 5. Additionally, this revision provides guidance for the quality group classification of components in non-LWR designs.

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, and metal containment components and their supports, and core support structures. These rules 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.

Historically, ASME developed and approved five Code Cases to address time-dependent material properties and failure modes. NRC RG 1.87, Revision 1, "Guidance for Construction of Class 1 Components in Elevated-Temperature Reactors (Supplement to ASME Section III Code Cases 1592, 1593, 1594, 1595, and 1596," issued June 1975 (Ref. 9), approved, with conditions, the initial versions of the five Code Cases (1592-0, 1593-0, 1594-0, 1595-0, and 1596-0). These Code Cases are the precursors to the other iterations of ASME's high temperature construction rules: Code Cases N-47 through N-51; ASME Code, Section III, Division 1, Subsection NH; and currently, ASME Code, Section III, Division 5. Except for the Code Cases reviewed in RG 1.87, Revision 1, and except for 10 CFR 50.55a (b)(1)(vi), the NRC has not formally reviewed or endorsed any of the other iterations of ASME's high temperature construction methods.

The NRC contracted with Pacific Northwest National Laboratory, Oak Ridge National Laboratory, Argonne National Laboratory, and NUMARK Associates, Inc. (NUMARK), to perform technical reviews of ASME Code, Section III, Division 5. NUREG-2245 documents the NRC staff's review of the 2017 Edition of ASME Code Section III, Division 5, and Code Cases N-861 and N-862, and

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 Tmax for Various Classes of Permitted Materials."

uses the recommendations in the following contractor reports, as well as the NRC staff's independent technical expertise, to form the basis for the findings.

- "Pacific Northwest National Laboratory Technical Input for the Nuclear Regulatory Commission Review of the 2017 Edition of ASME Section III, Division 5, 'High Temperature Reactors,'" issued September 2020. (Ref. 10).

- "Oak Ridge National Laboratory Technical Input for the Nuclear Regulatory Commission Review of the 2017 Edition of the ASME Boiler and Pressure Vessel Code, Section III, Division 5, 'High Temperature Reactors,'" issued August 2020. (Ref. 11).

- NUMARK Associates, Inc., "Technical Input for the U.S. Nuclear Regulatory Commission Review of the 2017 Edition of ASME Boiler and Pressure Vessel Code, Section III, Division 5, 'High-Temperature Reactors' HBB-T, HBB-II, HCB-I, HCB-II, and HCB-III for Metallic Components," issued December 2020. (Ref. 12).

- NUMARK Associates, Inc., "Technical Input for the U.S. Nuclear Regulatory Commission Review of the 2017 Edition of ASME Boiler and Pressure Vessel Code, Section III, Division 5, 'High Temperature Reactors': Subsection HH, "'Class A Nonmetallic Core Support Structures,'" Subpart A, "Graphite Materials," issued December 2020. (Ref. 13).

- NUMARK Associates, Inc., "Technical Input for the U.S. Nuclear Regulatory Commission Review of the 2017 Edition of ASME Boiler and Pressure Vessel Code, Section III, Division 5, 'High-Temperature Reactors.' Review of Code Case N-861 and N-862: Elastic-Perfect Plastic Methods for Satisfaction of Strain Limits and Creep-Fatigue Damage Evaluation in BPV-III-5 Rules," issued December 2020. (Ref. 14).

- Argonne National Laboratory, "Historical Context and Perspective on Allowable Stresses and Design Parameters in ASME Section III, Division 5, Subsection HB, Subpart B (ANL/AMD-21/1)," issued March 2021. (Ref. 15).

On August 20, 2021, the NRC published DG-1380 (proposed Revision 2 to this RG), "Acceptability of ASME Code Section III, Division 5, 'High Temperature Reactors,'" (Ref. 16), for public comment. Subsequent to the public comment period for DG-1380, the NRC staff completed its review of Code Cases N-872 and N-898, related to the use of Nickel-Based Alloy 617. On March 1, 2022, the NRC staff issued a supplemental *Federal Register* notice (Ref. 17) to DG-1380 requesting public comment on the staff's proposed endorsement of Code Cases N-872 and N-898. The technical basis for the NRC's endorsement of Code Cases N-872 and N-898 is contained in Technical Letter Report (TLR)-RES/DE/REB-2022-01, "Review of Code Cases Permitting Use of Nickel-Based Alloy 617 in Conjunction with ASME Section III, Division 5," dated January 31, 2022 (Ref. 18).

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-861, N-862, N-872, and N-898.

Appendix A to this RG provides guidance for the quality group classification of components in non-LWR designs. It provides one method that is acceptable to the NRC staff for the safety classification of components for non-LWR nuclear power plants. An 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.

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. 19) and Management Directive and Handbook 6.6, "Regulatory Guides" (Ref. 20).

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. 21).

Documents Discussed in Staff Regulatory Guidance

This RG endorses the use of one or more codes or standards developed by external organizations. These codes and standards 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 2017 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 2017 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 is also endorsing use of certain values in the 2019 Edition of ASME Code, Section II, Part D and Mandatory Appendix HBB-I-14 of the 2019 Edition of the ASME Code, Section III, Division 5 for limited use, as explained further in Condition C.1.u, below.

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.

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 as indicated in NUREG-2245, 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-3855.3, Approval and Control of Suppliers of Subcontracted Services

   When using HAB-3126(a), HAB-3127(a), and HAB-3855.3(c)(1) and (d)(1):

   (1) Accreditation should be in accordance with the 2017 edition of the International Organization for Standardization (ISO)/International Electrotechnical Commission (IEC) 17025, "General Requirements for the Competence of Testing and Calibration Laboratories," (Ref. 22) and should be from an accredited body recognized by the International Laboratory Accreditation Cooperation (ILAC) Mutual Recognition Arrangement (MRA).

   (2) 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-3855.3(c)(2) and (d)(2):

(3) The procurement documents should specify that the service will be provided in accordance with the accredited ISO/IEC 17025 program and scope of accreditation.

(4) 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.

When using HAB-3126(c)(1), HAB-3127(c)(1), and HAB-3855.3(c)(3) and (d)(3):

(5) At receipt inspection, the GC Certificate Holder or Graphite Material Organization should be responsible for confirming that the supplier's documentation certifies that the services (subcontracted calibration or testing, as applicable) were performed in accordance with the supplier's ISO/IEC 17025 program and scope of accreditation.

e. HAB-3220, Categories of the Owner's Responsibilities

(1) When using HAB-3220, the applicant or licensee should also apply the following provision from NCA-3220, "When the Owner[4] assigns any of the responsibilities listed in [NCA-3220] (e) through (u) above, such assignment shall contain, as a minimum, the name and address of the designee, the responsibilities being assigned, and the applicable nuclear facility or facilities," replacing the reference to NCA-3220(e) through (u) with HAB-3220(e) through (r).

f. HAB-3842.2, Evaluation of the Qualified Material Organization's Program by GC Certificate Holders

(1) When using HAB-3842.2(g), the applicant or licensee should also apply the provisions from NCA-3842.2(h) and NCA-3842.2(i), replacing the references to NCA-4259.1(a) through (c) with HAB-3859.1(a) through (e) and references to Material Organization with Graphite Material Organization.

g. HAB-3853.1, Quality System Manual

(1) When using HAB-3853.1, the applicant or licensee should also apply NCA-4253.1(d), replacing the reference to Material Organization with Graphite Material Organization.

[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."

[4] For purposes of this RG, the term "Owner" is not used in its the plain language meaning but rather used as defined in Subarticle NCA-9200, "Definitions," of ASME Code, Section III, Division 1 as follows. "Owner: organization legally responsible for the construction and/or operation of a nuclear facility including but not limited to one who has applied for, or has been granted, a construction permit or operating license by the regulatory authority having lawful jurisdiction."

h. HAB-3855.3, Approval and Control of Suppliers of Subcontracted Services

(1) When using HAB-3855.3(b), the applicant or licensee should also apply NCA-4255.3(b), replacing the reference to Material Organization with Graphite Material Organization and the reference to Certificate Holder to GC Certificate Holder.

i. HAB-4000, Quality Assurance

(1) When using HAB-4000, the applicant or licensee should also apply NCA-4133, replacing the reference to Material Organization with Graphite Material Organization, the reference to N type Certificate Holder with GC Certificate Holder, and the reference to Quality System Certificate with Graphite Quality System Certificate.

j. HAB-4134.6, Document Control

(1) When using HAB-4134.6, the applicant or licensee should also apply NCA-4134.6, replacing the reference to NCA with HAB and replacing the reference to Certificate Holder with GC Certificate Holder.

k. HAB-4134.7, Control of Purchased Items and Services

(1) When using HAB-4134.7, the applicant or licensee should also apply NCA-4134.7(e) and NCA-4134.7(g), replacing the reference to NCA with HAB, except that the provision that states "see NCA-4255.5 for unqualified source material" is not applicable to HAB-4134.7, and therefore the NRC staff does not endorse that provision for use in connection with HAB-4134.7.

l. HAB-5125, Duties of Authorized Nuclear Inspector Supervisor (Graphite)

(1) When using HAB-5125, the applicant or licensee should also apply NCA-5125(h) and (i), replacing the reference to Supervisor with Authorized Nuclear Inspector Supervisor (Graphite) and the reference to Certificate Holder to GC Certificate Holder.

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

(1) When using HAB-5230, the applicant or licensee should also apply NCA-5230(b), (c), and (d), replacing the reference to Inspector with Authorized Nuclear Inspector (Graphite), the references to NCA with HAB (except that NCA-3550 and NCA-3555 should be replaced with HAB-3450 and HAB-3455, respectively), and the references to Certificate Holder with GC Certificate Holder.

n. HAB-5000, Authorized Inspection

(1) When using HAB-5000, the applicant or licensee should also apply NCA-5256, Nondestructive Examination Personnel, replacing the reference to Inspector with Authorized Nuclear Inspection (Graphite), the reference to Material Organization with Graphite Material Organization, and the reference to Certificate Holder with GC Certificate Holder.

o. 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 Inspector with Authorized Nuclear Inspection (Graphite) and the reference to NCA with HAB.

p. HAB-7100, General Requirements

(1) Consistent with C.1.d above, and in addition to the references listed in Table HAB-7100-1, the applicant or licensee should also include ISO/IEC 17025, issued 2017, as the acceptable standard for use.

q. HAB-8180, Renewal

(1) When using HAB-8180, the applicant or licensee should also apply NCA-8182(a) and (b), replacing the reference to Authorized Nuclear Inspector Supervisor with Inspector Supervisor (Graphite).

r. HBB-3430, Pump Types

(1) The NRC staff does not endorse HBB-3430 as written. Instead, the applicant or licensee should use the following: Descriptions and definitions of common pump types are listed in NB-3440.

s. 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 Section III 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.

t. 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 t.(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).

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

(1) The NRC staff does not endorse the following materials properties in Mandatory Appendix HBB-I-14[5]

(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.

[5] 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 Section III-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 °F	Time (hr)										
	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 °C	Time (hr)										
	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 Section III-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) 9Cr-1Mo-V S_0, S_{mt}, S_t, and S_r values.

(e) 9Cr-1Mo-V R-factors in Table HBB-I-14.10E for temperatures greater than 525 °C (977 °F). [6]

(f) 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.[7]

(2) For 9Cr-1Mo-V, the NRC staff is endorsing the use of certain values in the 2019 Edition of Section II, Part D and Mandatory Appendix HBB-I-14 of the 2019 edition of ASME Code Section III, Division 5 in place of the values in the 2017 edition:

(a) S_0 values should be based on the larger of the S values in Section II, Part D (2019 Edition) and the S_{mt} values at 300,000 hours in Section III-5 Table HBB-I-14.3E (2019 edition).

(b) S_{mt} values should be based on the values in Table HBB-I-14.3E from the 2019 Edition of Section III-5.

(c) S_t values should be based on the values in Table HBB-I-14.4E from the 2019 Edition of Section III-5.

(d) S_r values should be based on the values in Table HBB-I-14.6F from the 2019 Edition of Section III-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.”

(b) Under Note (6) clause (c), “Note (4)” should be changed to “Note (5).”

(c) In the line for SA 234, “WP22, WP22W” should be replaced with “WP22 CL1, CL3.”

(d) For base material Type 304 SS and Type 316 SS, for Specification SA 403, Grades WP 304W, WP 304HW, WP 316W, and WP 316HW should be removed from the list of grades.

v. 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

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6 Unless ASME approves and the NRC endorses the proposed R-factors in ASME Code Record 17-2817, the NRC will evaluate applications to use them on a case-by-case basis with appropriate justification.

7 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.

constitutive models used in assessments for cyclic creep loading. The validity of the inelastic constitutive models should be demonstrated in the design report.

w. 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.

x. 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.

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

(1) The NRC staff did not review Nonmandatory Appendix HBB-Y and therefore is not endorsing it.

z. 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.

aa. HHA-3141, Oxidation

(1) The NRC staff is not endorsing the provisions of HHA-3141(c) that set the weight loss limit as 30 percent for geometry reduction in the oxidation analysis. Designers[8] 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 oxidation analysis.

bb. 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.

cc. HHA-3143, Abrasion and Erosion

[8] For purposes of this RG, the term "Designer" 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. "Designer: the organization responsible for preparation of Design Output Documents."

(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.

dd. HHA-3330, Design of the Graphite Core Assembly

(1) The NRC staff is not endorsing the provisions of HHA-3330(g) because provisions for inservice inspection are outside of the scope of ASME Code, Section III, Division 5.

ee. 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.

ff. 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.

gg. 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 do not provide guidance for applicants or licensees.

2. Section III, Division 5, Code Cases

The NRC staff endorses the Code Case 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	SUPPLEMENT/ EDITION[9]
N-872	Use of 52Ni-22Cr-13Co-9Mo Alloy 617 (UNS N06617) for Low Temperature Service Construction, Section III,Division 5.”	0/17E

[9] The third column of Tables 4 lists the supplement and edition in which each Code Case was published (e.g., “5/15E” means Supplement 5 to the 2015 Edition of the ASME BPV Code).

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 as indicated in NUREG-2245. The NRC staff is unable to review those sections identified as in the course of preparation to determine whether or not they are acceptable, and therefore the staff does not endorse them.

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

CODE CASE NUMBER	CODE CASE TITLE/LIMITATION	SUPPLEMENT/ EDITION[10]
N-861	*Satisfaction of Strain Limits for Division 5 Class A Components at Elevated Temperature Service Using Elastic-Perfectly Plastic Analysis*	5/15E
	When using subarticle 5.3, the applicant or licensee should refer to Table HBB-I-14.1(b), "Permissible Weld Materials," in place of Table HBB-I-14.10.	
N-862	*Calculation of Creep-Fatigue for Division 5 Class A Components at Elevated Temperature Service Using Elastic-Perfectly Plastic Analysis*	5/15E
	When using subarticle 6.4, the applicant or licensee should refer to Table HBB-I-14.1(b), "Permissible Weld Materials," in place of Table HBB-I-14.10.	
N-898	*Use of Alloy 617 (UNS N06617) for Class A Elevated Temperature Service Construction Section III, Division 5*	4/19E
	When applying HBB-T-1710 and HBB-4800 to Alloy 617 components, applicants and licensees should develop their own plans to address the potential for stress relaxation cracking in their designs. These plans should address factors such as weld joint design and controls on welding in addition to the required heat treatment of HBB-4800. When applying HBB-T-1836(2)(-b), the equation for plastic strain in the code case should be replaced with the following equation: $for\ \sigma > \sigma_1, \varepsilon_p = -\frac{1}{\delta} ln\left(1 - \frac{\sigma - \sigma_1}{\sigma_p - \sigma_1}\right)$	

10 The third column of Table 5 lists the supplement and edition in which each Code Case was published (e.g., "5/15E" means Supplement 5 to the 2015 Edition of the ASME BPV Code).

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. 23), 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 [11]

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," 2017 Edition, New York, NY.[12]

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. ASME BPV Code, Section II, "Materials," Part D, "Properties (Metric)," 2019 Edition, New York, NY

8. ASME BPV Code, Section III, Division 1, "Rules for Construction of Nuclear Facility Components," 2017 Edition, New York, NY

9. NRC RG 1.87, Revision 1, "Guidance for Construction of Class 1 Components in Elevated-Temperature Reactors (Supplement to ASME Section III Code Cases 1592, 1593, 1594, 1595, and 1596," Washington, DC, June 1975. (ML003740252)

10. Pacific Northwest National Laboratory, et al., "Pacific Northwest National Laboratory Technical Input for the Nuclear Regulatory Commission Review of the 2017 Edition of ASME Section III, Division 5, 'High Temperature Reactors,'" with Engineering Mechanics Corporation of Columbus, September 2020. (ML20269A145)

11. Oak Ridge National Laboratory, "Oak Ridge National Laboratory Technical Input for the Nuclear Regulatory Commission Review of the 2017 Edition of the ASME Boiler and Pressure Vessel Code, Section III, Division 5, 'High Temperature Reactors,'" with Clarus Consulting, LLC, and

11 Publicly available NRC published documents are available electronically through the NRC Library on the NRC's public Web site 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. The documents can also be viewed online or printed for a fee in the NRC's Public Document Room (PDR) at 11555 Rockville Pike, Rockville, MD. For problems with ADAMS, contact the PDR staff at 301-415-4737 or (800) 397-4209; fax (301) 415-3548; or e-mail pdr.resource@nrc.gov.

12 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.

RWH consult GmbH Oberrohrdorf, Technical Letter Report ORNL/SPR-2020/1653, August 2020. (ML20269A125)

12. NUMARK Associates, Inc., "Technical Input for the U.S. Nuclear Regulatory Commission Review of the 2017 Edition of ASME Boiler and Pressure Vessel Code, Section III, Division 5, 'High-Temperature Reactors,' HBB-T, HBB-II, HCB-I, HCB-II, and HCB-III for Metallic Components," with Engineering Mechanics Corporation of Columbus, Technical Letter Report TLR/RES/DE/CIB-2020-13, December 2020. (ML20349A003)

13. NUMARK Associates, Inc., "Technical Input for the U.S. Nuclear Regulatory Commission Review of the 2017 Edition of ASME Boiler and Pressure Vessel Code, Section III, Division 5, 'High Temperature Reactors': Subsection HH, 'Class A Nonmetallic Core Support Structures,' Subpart A, 'Graphite Materials,'" Technical Letter Report TLR/RES/DE/CIB-2020-10, December 2020. (ML20344A001)

14. NUMARK Associates, Inc., "Technical Input for the U.S. Nuclear Regulatory Commission Review of the 2017 Edition of ASME Boiler and Pressure Vessel Code, Section III, Division 5, 'High-Temperature Reactors.' Review of Code Case N-861 and N-862: Elastic-Perfect Plastic Methods for Satisfaction of Strain Limits and Creep-Fatigue Damage Evaluation in BPV-III-5 Rules," with Engineering Mechanics Corporation of Columbus, Technical Letter Report TLR/RES/DE/CIB-2020-14, December 2020. (ML20349A002)

15. Argonne National Laboratory, "Historical Context and Perspective on Allowable Stresses and Design Parameters in ASME Section III, Division 5, Subsection HB, Subpart B" (ANL/AMD-21/1), March 2021. (ML21090A033)

16. NRC, Draft Guide 1380, "Acceptability of ASME Code Section III, Division 5, High Temperature Reactors," Washington, DC, August 2021. (ML21091A276).

17. NRC, "Acceptability of ASME Code Section III, Division 5, High Temperature Reactors," *Federal Register*, Vol. 87, No. 40, March 1, 2022, pp. 11490–11492.

18. NRC, Technical Letter Report (TLR)-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 31, 2022. (ML22031A137).

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

20. NRC, Management Directive (MD) 6.6, "Regulatory Guides," Washington, DC, May 2, 2016. (ML18073A170)

21. 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.[13]

[13] 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.

22. International Organization for Standardization (ISO)/ International Electrotechnical Commission (IEC), ISO/IEC 17025:2017, "General Requirements for the Competence of Testing and Calibration Laboratories," Geneva, Switzerland, 2017.[14]

23. NRC, MD 8.4, "Management of Backfitting, Forward Fitting, Issue Finality, and Information Requests," Washington, DC.

[14] Copies of International Organization for Standardization (ISO) standards may be purchased from the ISO Web site (https://www.iso.org/store.html).

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 Regulatory Guide (RG) 1.233, "Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors," (Ref. A-3). The guidance in this appendix establishes quality group assignments of mechanical systems and components of non-light-water reactors acceptable to the U.S. Nuclear Regulatory Commission (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.

American Society of Mechanical Engineers (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 safety-related (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 light-water

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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

reactors (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 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.

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 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, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants

[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.

(Ref. A-15). The NRC staff has determined that ASME 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 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

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.

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 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

[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.

- 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 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.

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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.

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

Classification Method	Component Classification		
Traditional	**Quality Group A**	**Quality Group B**	**Quality Group C**
Risk-Informed (10 CFR 50.69)[6]	**RISC-1**	**RISC-1**	**RISC-2, RISC-3**
Risk-Informed (RG 1.233)	**SR**	**SR**	**NSRST**
	SR Quality Design Standards		**Important to Safety Design Standards**
Components			
Pressure Vessels	ASME Code, Section III, Division 5, Class A	ASME Code, Section III, Division 5, Class B	ASME Code, Section VIII, Division 1 or Division 2[7]
Piping			ASME B31.1/B31.3[7]
Pumps			
Valves			ASME B31.1/B31.3[7]
Atmospheric Storage Tanks			
Storage Tanks (0-15 pounds per square inch gauge)			ASME Code, Section VIII, Division 1 or Division 2[7]
Metallic Core Support Structures	ASME Code, Section III, Division 5, Subsection HG	N/A	
Nonmetallic Core Support Structures	ASME Code, Section III, Division 5, Subsection HH	N/A	

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 address design in high temperature environments and may be acceptable with appropriate justification. Applicants may propose alternate standards with appropriate justification.

APPENDIX A REFERENCES[8]

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).[9]

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," July 1998 Edition with July 1999 Addenda, New York, NY.

A-9. ASME, Boiler and Pressure Vessel Code, Section VIII, Division 2, "Alternative Rules," July 1998 Edition with July 1999 Addenda Edition, New York, NY

A-10. ASME Standard B31.1, "Power Piping," 1999 Edition, New York, NY.

A-11. ASME Standard B31.3, "Process Piping," 1999 Edition, New York, NY.

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

8 Publicly available NRC published documents are available electronically through the NRC Library on the NRC's public Web site 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. The documents can also be viewed online or printed for a fee in the NRC's Public Document Room (PDR) at 11555 Rockville Pike, Rockville, MD. For problems with ADAMS, contact the PDR staff at 301-415-4737 or (800) 397-4209; fax (301) 415-3548; or e-mail pdr.resource@nrc.gov.

9 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-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.[10]

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).

[10] 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.