

U.S. NUCLEAR REGULATORY COMMISSION

DRAFT REGULATORY GUIDE DG-1380



Proposed Revision 2 to Regulatory Guide 1.87

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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).

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.

This RG is being issued in draft form to involve the public in the development of regulatory guidance in this area. It has not received final staff review or approval and does not represent an NRC final staff position. Public comments are being solicited on this DG and its associated regulatory analysis. Comments should be accompanied by appropriate supporting data. Comments may be submitted through the Federal rulemaking Web site, <http://www.regulations.gov>, by searching for draft regulatory guide DG-1380. Alternatively, comments may be submitted to the Office of Administration, Mailstop: TWFN 7A-06M, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, ATTN: Program Management, Announcements and Editing Staff. Comments must be submitted by the date indicated in the *Federal Register* notice.

Electronic copies of this DG, previous versions of DGs, and other recently issued guides are available through the NRC’s public Web site under the Regulatory Guides document collection of the NRC Library at <https://nrcweb.nrc.gov/reading-rm/doc-collections/reg-guides/>. The DG is also available through the NRC’s Agencywide Documents Access and Management System (ADAMS) at <http://www.nrc.gov/reading-rm/adams.html>, under Accession No. ML21091A276. The regulatory analysis may be found in ADAMS under Accession No. ML21091A277.

- 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 associated 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,” states, in part, that structures, systems, and components 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.

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

- 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 structures, systems, and components 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.

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), approval 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 OMB reviewer at: OMB Office of Information and Regulatory Affairs (3150-0011 and 3150-0151), Attn: Desk Officer for the Nuclear Regulatory Commission, 725 17th Street, NW Washington, DC20503; e-mail: oira_submission@omb.eop.gov.

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 and N-862 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 temperature² 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 rules.

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 associated Code Cases N-861 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.”

N-862, and 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).
- “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).
- “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).
- “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).
- “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).

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 and N-862.

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. 16) and Management Directive and Handbook 6.6, “Regulatory Guides” (Ref. 17).

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. 18).

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 that use of Division 1 for high-temperature reactors, with the exceptions and limitations stated below. An applicant who wishes to follow the guidance in this regulatory guide 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. 19) 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 services³ 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 Owner⁴ 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

- (a) Type 304 stainless steel (Type 304 SS) values of S_{mt} , S_t , and S_r for temperatures greater than 1300 °F or 700 °C.
 - (b) Type 316 stainless steel (Type 316 SS) S_r values for temperatures greater than 1300 °F or 700 °C.
 - (c) 2-1/4Cr-1Mo material S_{mt} , S_t , and S_r values for temperatures greater than 950 °F or 510 °C.
 - (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).⁵
 - (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.⁶
- (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(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.”

⁵ 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.

⁶ 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.

- (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 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) that 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. HGB-3224, Level C Service Limits
- (1) When extrapolating t_{ib} using Figures HBB-I-14.4A through HBB-I-14.4E to obtain t_{ib} in accordance with HGB-3224(d), the maximum t_{ib} value for any stress and temperature combination should not exceed 300,000 hours or the end of the curve for the temperature of interest, whichever is less.
- aa. 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) that significant elastic follow-up is not occurring.
- bb. 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. Designers⁷ should determine

⁷ 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.”

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.

cc. 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.

dd. 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.

ee. 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.

ff. 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.

gg. 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.

hh. 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 Cases listed in Table 1 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. The third column of Table 1 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).

Table 1. Conditionally Acceptable ASME Code, Section III, Division 5 Code Cases

CODE CASE NUMBER	CODE CASE TITLE/EXCEPTION	SUPPLEMENT/ EDITION
N-861	<i>Satisfaction of Strain Limits for Division 5 Class A Components at Elevated Temperature Service Using Elastic-Perfectly Plastic Analysis</i>	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	<i>Calculation of Creep-Fatigue for Division 5 Class A Components at Elevated Temperature Service Using Elastic-Perfectly Plastic Analysis</i>	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.	

D. IMPLEMENTATION

The NRC staff may use this regulatory guide 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 regulatory guide 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. 20), 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 regulatory guide 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.

REFERENCES⁸

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.⁹
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. ML21223A097)
5. NRC, Regulatory Guide (RG) 1.232, Revision 0, “Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors,” Washington, DC. (ADAMS Accession No. 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. (ADAMS Accession No 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. (ADAMS Accession No. 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. (ADAMS Accession No. 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

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 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. (ADAMS Accession No. 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. (ADAMS Accession No. 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. (ADAMS Accession No. 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. (ADAMS Accession No. ML20349A002)
15. 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. (ADAMS Accession No. ML21090A033)
16. NRC, “Nuclear Regulatory Commission International Policy Statement,” *Federal Register*, Vol. 79, No. 132, July 10, 2014, pp. 39415–39418.
17. NRC, Management Directive (MD) 6.6, “Regulatory Guides,” Washington, DC, May 2, 2016. (ADAMS Accession No. ML18073A170)
18. 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, 2016.¹⁰
19. 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, 2017.¹¹
20. NRC, MD 8.4, “Management of Backfitting, Forward Fitting, Issue Finality, and Information Requests,” Washington, DC.

¹⁰ 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.

¹¹ 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 approach outlined 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 the standards established in this appendix on a case-by-case basis to determine if the proposals are appropriate for the specific design.

A-2. Safety Classification Categories

Traditional Approach

In the traditional approach outlined in 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. Under the traditional approach for light-water 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-4), 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 American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) Code (ASME Code) classification system. Quality Group A is aligned with ASME Class A, and consequently Class 1 for reactor coolant system pressure boundary components; Quality Group B is aligned with ASME Class B, and consequently 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 C, and consequently Class 3.

SSCs that are NSR may have some special function, such as preventing a radiological release to the public by ensuring that no dose to the public is beyond the regulatory limits of 0.1 rem total effective dose equivalent (TEDE) set by 10 CFR Part 20 “Domestic Licensing of Production and Utilization Facilities,” (Ref. A-5). While such SSCs do not meet the criteria for an SR SSC, there is still a need to ensure component integrity. These SSCs are NSR, but are considered to have special treatment, so the use of ASME Code, Section VIII, Division 1, “Rules for Construction of Pressure Vessels,” (Ref. A-6), ASME Code, Section VIII, Division 2, “Alternative Rules,” (Ref. A-7), and piping codes such as ASME B31.1, “Power Piping” (Ref. A-8), or ASME B31.3, “Process Piping,” (Ref. A-9), is appropriate to ensure their integrity. RG 1.26 and RG 1.143 endorse the use of ASME Code, Section VIII, B31.1, and B31.3, as described here.

RG 1.26 assigns Quality Group D to other components that contain or may contain radioactivity but are not part of the reactor coolant pressure boundary or included in Quality Groups B or C. Refer to RG 1.26¹ for more information on this traditional approach. RG 1.143 “Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants” (Ref. A-10),² provides information related to the classification of radioactive waste management systems that fall within the scope of that RG. 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.

In the transition to the establishment of quality groups for non-LWRs, the high-temperature design rules for mechanical components in the ASME Code, Section III, Division 5, “High Temperature Reactors” (Ref. A-11) uses only two classes, Class A and Class B. 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 only when the consequences of a failure are less significant. The differences in the classes involve how creep and thermal cycling are treated.

For SR mechanical SSCs, the ASME Code, Section III should be used. Both Division 1, “Rules for Construction of Nuclear Facility Components” (Ref. A-12) 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 non-LWRs that operate below these temperatures, Division 5, which refers to the rules in Division 1, should be followed. NSR mechanical components that need special treatment, such as for systems containing high levels of radioactive material, should be designed to the ASME Code, Section VIII, Division 1 or 2 for vessels and ASME B31.1 or ASME B31.3 for piping depending on whether the components are power piping or process piping.

¹ RG 1.26 has used a 500 millirem whole body dose criteria since Revision 1 (1974), which was the annual public dose limit at that time. As of the development of this Appendix, the NRC is updating RG 1.26 to reflect the current public dose limit of 100 millirem TEDE.

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-13) to the 100 millirem criterion.

Risk-Informed Approach

In 10 CFR 50.69, the NRC provides an alternative classification system that establishes four categories for SSCs. 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 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 (Ref. A-15). Class A should be used for SR SSCs that perform safety significant functions (RISC-1). 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 these purposes, are appropriate standards to apply to NSR SSCs that need special treatment. SR SSCs that perform low safety significant functions are designated as SR and should be designed, fabricated, erected, and tested under a quality assurance program that meets the requirements in 10 CFR Part 50, Appendix B. SR mechanical SSCs should be designed to ASME Code, Section III standards, and NSR mechanical SSCs may be designed to other standards. For high-temperature applications, ASME Code, Section III, Division 5 is appropriate. SR SSCs that perform low safety significant functions (RISC-3) should use at a minimum ASME Code, Section III, Division 5, Class B. NSR SSCs that perform low safety significant functions (RISC-4) are left to the designers and owners to establish the applicable design standards, as these SSCs are generally needed to support commercial aspects of the facility.

License Modernization Process 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 systems, structures, and components (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 18-04³ gives the following definitions for these terms:

³ 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 are expected to use the terminology in NEI 18-04 and, as needed, identify exceptions to and exemptions needed from NRC regulations.

- 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
- 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)

In this system of classification, 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 low safety-significant functions. NSRST may be those systems that are relied on for defense in depth or ensure that releases to the public from systems containing radioactive material could result in doses to the public exceeding 0.1 rem. Acceptable standards for NSR SSCs with special treatment are ASME Code, Section VIII, along with ASME B31.1 or B31.3 piping, which RG 1.26 and RG 1.143 endorse for these purposes. NSR components with no special treatment are left to the designers and owners to establish applicable standards.

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. 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 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 NB 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 NC. 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. Appropriate standards to use for NSR SSCs that warrant special treatment for functions such as defense-in-depth or to maintain exposure to the public from failures in rad-waste-containing systems within regulatory limits are those found in ASME Code, Section VIII, Division 1 or 2, along with piping codes ASME B31.1 or ASME B31.3 as applicable. 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, four quality groups are established. The quality groups are based on the classification of the SSC as either SR or NSR, and then based on the safety significance of the SSC. These quality groups correspond to the four categories in the risk-informed approach of 10 CFR 50.69. SSCs classified as NSR are separated depending on special treatment needs for the SSC based on SSC function. Assignment of appropriate design standards to the SR classification will depend upon the consequence of component failure. This will dictate the appropriate design class for use. The NRC would expect core support SSCs to be classified as SR because these components ensure the core configuration is maintained in an analyzed configuration.

NEI 18-04 describes a process that may be used to determine the safety classification of SSCs. Once that process is completed, Table A-1, "Classification and Standards Applicable to Advanced Reactors," may be used to identify an appropriate standard for the design and fabrication of an SSC. The traditional approach is based on the definition of an SR SSC in 10 CFR 50.2. The process outlined in 10 CFR 50.69 starts with the definition of SR in 10 CFR 50.2 but then applies probabilistic analysis to determine the importance of the SSC with regard to safety significance, either high or low. At the end of the process, the four categories for component classification result. Once these are determined, Table A-1 may be used to identify an appropriate standard for the design and fabrication of the SSC.

If the traditional approach based on the definition of SR SSC in 10 CFR 50.2 is used, the considerations below may be used to determine if the SSC is SR, NSRST, or NSR.

Quality Group A: SR SSCs that meet the 10 CFR 50.2 definition of safety related but have a high safety significance from failure (RISC-1).

- SSCs that are relied on to remain functional during and following design basis events to assure the following:
 - the integrity of the reactor coolant boundary
 - the capability to shut down the reactor and maintain it in a safe-shutdown condition
 - the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guideline exposures set forth in 10 CFR 50.34(a)(1) or 10 CFR 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Centre Distance," as applicable
- Quality Group A SR SSCs would include those components determined to be seismic Category I

Quality Group B: SR SSCs that meet the 10 CFR 50.2 definition of safety related but have a low safety significance from failure (RISC-3).

- Quality Group B SR SSCs could include those components determined to be seismic Category I, whose only SR function is to provide support for other SR components, if failure would result in low safety significance.

Quality Group C: NSRST SSCs that do not meet the definition of 10 CFR 50.2 for safety related but perform a defense-in-depth function, contain radioactivity, or protect the health and safety of the public from a radiological release (RISC-2).

Possible SSCs for such consideration include the following:

- components for which structural integrity is needed after a seismic event to ensure the function of a Quality Group A component is not impacted
- components used for defense-in-depth purposes
- components whose failure could create conditions that may impact Quality Group A components (e.g., the Quality Group B component is a secondary heat transfer loop, the failure of which could result in pressurization of the primary system beyond the design pressure or could impact other Quality Group A SSCs)
- those SSCs that may contain radioactivity during routine normal operation or from a routine operating occurrence

Quality Group D: NST

- SSCs that do not meet the criteria for Quality Group A, B, or C (RISC-4)

Table A-1. Classification and Standards Applicable to Advanced Reactors for Quality Groups A, B, C, and D

Components	QUALITY STANDARDS			
	Quality Group A (SR Classification)	Quality Group B (SR Classification)	Quality Group C (NSR with Special Treatment Classification)	Quality Group D (NSR with No Special Treatment Classification)
	RISC-1	RISC-3	RISC-2	RISC-4
	SR	SR	NSRST	NST
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 ASME Code, Section VIII, Division 2	Owner Defined
Piping			ASME B31.1/B31.3	Owner Defined
Pumps			Manufacturers' standards	Owner Defined
Valves			ASME B31.1/B31.3	Owner Defined
Atmospheric Storage Tanks			ASME Code, Section VIII, Division 1, ASME Code, Section VIII, Division 2, or Manufacturers' standards	Owner Defined
Storage Tanks (0-15 pounds per square inch gauge)			ASME Code, Section VIII, Division 1, ASME Code, Section VIII, Division 2, or Manufacturers' standards	Owner Defined
Core Support Structures	ASME Code, Section III, Division 5, Subsection HG/HH	ASME Code, Section III, Division 5, Subsection HG/HH	N/A	Owner Defined

APPENDIX A REFERENCES⁴

- 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, August 2019. (ADAMS Accession No, ML19241A472).⁵
- 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 (ADAMS Accession No. ML20091L698).
- A-4. 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 (ADAMS Accession No ML16286A590).
- A-5. 10 CFR Part 20, “Domestic Licensing of Production and Utilization Facilities.”
- A-6. 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.
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⁴ 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.

⁵ 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. 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.⁶
- A-13. NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” Washington, DC.
- 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 (ADAMS Accession No. 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 (ADAMS Accession No. ML17325A611).

⁶ 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>.