U.S. NUCLEAR REGULATORY COMMISSION

REGULATORY GUIDE 1.26, REVISION 6

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QUALITY GROUP CLASSIFICATIONS AND STANDARDS FOR WATER-, STEAM-, AND RADIOACTIVE-WASTE-CONTAINING COMPONENTS OF NUCLEAR POWER PLANTS

A. INTRODUCTION

Purpose

This regulatory guide (RG) describes a quality classification system related to specified national standards that may be used to determine quality standards acceptable to the staff of the U.S. Nuclear Regulatory Commission (NRC) for satisfying General Design Criterion (GDC) 1, "Quality Standards and Records," of Appendix A, "General Design Criteria for Nuclear Power Plants," to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities" (Ref. 1), for components containing water, steam, or radioactive material in light-water-cooled nuclear power plants. The appendix to this RG provides guidance for alternative quality classification systems for components in light-water reactor (LWR) nuclear power plants.

Applicability

This RG applies to all holders of and applicants for construction permits and operating licenses for nuclear power reactors subject to the provisions of 10 CFR Part 50, Appendix A, including those that have permanently ceased operations and have certified that fuel has been permanently removed from the reactor vessel, and all holders of and applicants for a power reactor combined license, design certification, standard design approval, or manufacturing license under 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants" (Ref. 2).

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://nrcweb.nrc.gov/reading-rm/doc-collections/reg-guides/, under Document Collections, in Regulatory Guides, at https://nrcweb.nrc.gov/reading-rm/doc-collections/reg-guides/contactus.html.

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://nrcweb.nrc.gov/reading-rm/doc-collections/reg-guides/, 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.) ML21232A142. The regulatory analysis is associated with a rulemaking and may be found in ADAMS under Accession No. ML20168A893. The associated draft guide DG-1371, Revision 0, may be found in ADAMS under Accession No. ML20168A883, and the staff responses to the public comments on DG-1371 may be found under ADAMS Accession No. ML17123A319.

Applicable Regulations

- 10 CFR Part 50 provides regulations for licensing production and utilization facilities.

  - 10 CFR Part 50, Appendix A, GDC 1, requires that structures, systems, and components (SSCs) important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. This RG provides guidance on the quality standards appropriate to various classes of SSCs.

  - 10 CFR 50.54, "Conditions of licenses," paragraph (jj), requires structures, systems, and components subject to the codes and standards in 10 CFR 50.55a must be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed.

  - 10 CFR 50.55, "Conditions of construction permits, early site permits, combined licenses, and manufacturing licenses," paragraph (i), requires structures, systems, and components.

  - Subject to the codes and standards in 10 CFR 50.55a must be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed.

  - 10 CFR 50.55a(c), 10 CFR 50.55a(d), and 10 CFR 50.55a(e) state that certain systems and components of boiling- and pressurized-water-cooled nuclear power reactors must be designed, fabricated, erected, and tested in accordance with the standards for Class 1, 2, and 3[1] components given in Section III, "Rules for Construction of Nuclear Facility Components," of the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (BPV Code) (Ref. 3) or equivalent quality standards. Footnote 7 to 10 CFR 50.55a, "Codes and Standards," references this as the guidance for quality group classifications that are to be included in safety analysis reports. In addition, the ASME *Operation and Maintenance of Nuclear Power Plants* (OM Code), Division 1, "Section IST: Rules for Inservice Testing of Light-Water Reactor Power Plants" (Ref. 4), covers an equipment scope that is further described in Section B below. This code is incorporated by reference in 10 CFR 50.55a.

  - 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors," provides a voluntary, risk-informed process for categorizing and treating (e.g., inspecting and testing) SSCs that may be used as part of an alternative to the process described in this RG. Appendix A to this RG provides additional detail.

Related Guidance

- NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (Ref. 5).

[1] In editions of the ASME BPV Code published before 1971, Section III uses the terms Class A, Class B, and Class C in lieu of Class 1, Class 2, and Class 3.

- o NUREG-0800, Section 3.2.2, "System Quality Group Classification," provides guidance to the NRC staff for reviewing the quality group classification of SSCs for nuclear power plant applications.

- o NUREG-0800, Section 17.5, "Quality Assurance Program Description—Design Certification, Early Site Permit, and New License Applicants," provides expectations for new reactor applicants related to the assurance of quality for certain nonsafety-related SSCs.

- RG 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants" (Ref. 6), provides specific guidance on the classification of radioactive waste management systems.

- RG 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance" (Ref. 7), provides guidance for complying with the NRC's voluntary requirements in 10 CFR 50.69, which is described further in Section B as well as in Appendix A to this RG.

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. 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, DC 20503; 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 Revision

This revision of the RG (Revision 6) includes an appendix that provides guidance for alternative quality classification systems for components in LWR nuclear power plants. In addition, this revision updates the staff position on the classification of Quality Group C components to incorporate current public radiation dose criteria and reflect the latest guidance on systems that contain radioactive material.

Background

In the early 1970s, the NRC staff developed a quality classification system to provide licensees with guidance for satisfying GDC 1. The system consists of four quality groups, A through D; methods for assigning components to those quality groups; and specific quality standards applied to each quality group. When the NRC issued Revisions 2 and 3 of this RG, 10 CFR 50.55a required that only components of the reactor coolant pressure boundary be designed, fabricated, erected, and tested to the highest available national standards; this corresponded to the quality standards for Quality Group A of the NRC system. The NRC issued Revision 4 to align this RG with the NRC regulations (Ref. 8), published on March 15, 1984, amending 10 CFR 50.55a to incorporate by reference the criteria in Section III of the ASME BPV Code as they relate to the design and fabrication of Class 2 and Class 3 components (Quality Group B and Group C components, respectively). Revision 5 of this RG provided better correlation between wording differences that exist between the ASME BPV Code and the NRC regulations not addressed in Revision 4. In addition, Revision 5 of the RG clarified content (e.g., the definition of Quality Group A and the scope of the ASME OM Code), corrected errors (e.g., a misplaced footnote), and provided additional references to related classification systems such as 10 CFR 50.69 and industry and international standards that applicants or licensees may propose as an alternative means to comply with NRC requirements.

The NRC Advisory Committee on Reactor Safeguards, in an October 17, 2016, letter describing the results of its review of Revision 5 of RG 1.26 (Ref. 9), recommended that in the next revision, RG 1.26 should be broadened to include a set of basic principles for assignment of components to each quality group. In a letter dated December 13, 2016 (Ref. 10), the NRC Executive Director for Operations described the planned NRC staff actions in response to this recommendation. Revision 6 of this RG provides more specific guidance for alternative quality classification systems for components in nuclear power plants in Appendix A to this RG.

Because the current quality group classification system for LWRs is well established, this revision of RG 1.26 retains the method described in previous versions for determining acceptable quality standards for Quality Group B, C, and D components. Other systems not covered by this RG, such as instrument and service air, diesel engines and 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. The evaluation to establish the quality group classification of these other systems should consider the guidance provided for Quality Groups B, C, and D in Sections C.1, C.2, and C.3 of this RG.

Key Safety Principles

The following paragraphs provide summary information and references related to other systems of classification and treatment that may be useful in applying the alternatives described in the appendix to this RG.

The American Nuclear Society (ANS) has prepared American National Standards Institute (ANSI)/ANS-58.14-2011, "Safety and Pressure Integrity Classification Criteria for Light Water Reactors" (Ref. 11), to provide criteria for the safety classification of items in LWR nuclear power plants, and for the assignment of pressure integrity classes to pressure-retaining items. ANSI/ANS-58.14-2011 denotes the categories with respect to safety classification using the terms "safety-related," "non-safety-related with augmented requirements," and "non-safety-related," and by Classes C-1 through C-5 for pressure integrity classification. As indicated in Footnote 39 in ANSI/ANS-58.14-2011, ANS Classes C-1 through C-4 generally correspond to Quality Groups A through D, respectively, as defined in 10 CFR 50.55a and RG 1.26. As described in Appendix A to this RG, use of this alternative classification approach instead of the approach presented in this RG is acceptable if an applicant or licensee provides sufficient basis and information for the NRC staff to verify that the proposed alternative demonstrates compliance with GDC 1 and 10 CFR 50.55a.

As mentioned above, the ASME OM Code is incorporated into the NRC's requirements in 10 CFR 50.55a. The OM Code specifies that its scope includes (1) pumps and valves that are required to perform a specific function in shutting down a reactor to the safe-shutdown condition, in maintaining the safe-shutdown condition, or in mitigating the consequences of an accident, (2) pressure relief devices that protect systems or portions of systems that perform one or more of the three functions identified in (1) above, and (3) dynamic restraints (snubbers) used in systems that perform one or more of the three functions identified in (1), or to ensure the integrity of the reactor coolant pressure boundary. A user of RG 1.26 should confirm that its classification process considers the scope of pumps, valves, and dynamic restraints specified in ASME OM Code.

Additionally, RG 1.201 provides guidance for complying with the NRC's voluntary requirements in 10 CFR 50.69. RG 1.201 specifies regulatory positions for the acceptable use of the process described in Nuclear Energy Institute (NEI) 00-04, "10 CFR 50.69 SSC Categorization Guideline," issued July 2005 (Ref. 12), to determine the safety significance of SSCs and place them into the appropriate risk-informed safety class categories. Through this process, the safety significance of SSCs is determined using an integrated decisionmaking process, which incorporates both risk and traditional engineering insights. The process considers the safety functions of SSCs to include both the design-basis functions (derived from the safety-related definition) and functions credited for preventing or mitigating severe accidents. An applicant or licensee may request the application of 10 CFR 50.69 as part of an alternative classification system for components in its nuclear power plant, as described in Appendix A to RG 1.201.

An applicant or licensee may request the application of 10 CFR 50.69 as part of an alternative classification system for components in its nuclear power plant, as 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 considered IAEA Safety Requirements and Safety Guides[2] pursuant

[2] IAEA Safety Requirements and Guides may be found at WWW.IAEA.Org/ or by writing the International Atomic Energy Agency, P.O. Box 100 Wagramer Strasse 5, A-1400 Vienna, Austria; telephone (+431) 2600-0; fax (+431) 2600-7; or e-mail Official.Mail@IAEA.Org. It should be noted that some of the international recommendations do not correspond to the requirements specified in the NRC's regulations and the NRC's requirements take precedence over the international guidance.

to the Commission's International Policy Statement and Management Directive and Handbook 6.6 (Ref. 13).

Although the NRC does not endorse IAEA Specific Safety Guide (SSG)-30, "Safety Classification of Structures, Systems and Components in Nuclear Power Plants" (Ref. 14), this RG generally incorporates similar guidelines and is generally consistent with the basic safety principles provided in it. Specifically, SSG-30 provides high-level guidance for developing safety categories and safety classes of SSCs in nuclear power plants with the use of probabilistic risk assessment considerations in the safety classification process.

As described below in Section D, the NRC staff may deem acceptable the use of this alternative classification approach instead of the approach presented in this RG, if an applicant or a licensee provides sufficient basis and information for the staff to verify that the proposed alternative complies with GDC 1 and 10 CFR 50.55a.

Documents Discussed in Staff Regulatory Guidance

This RG endorses, in part, the use of one or more codes or standards developed by external organizations, and other third party guidance documents, some of which (e.g., portions of the ASME BPV Code) have been incorporated by reference into NRC regulations as a requirement. These codes, standards and third party guidance documents may contain references to other codes, standards or third party guidance documents ("secondary references"). If a secondary reference has itself been incorporated by reference into NRC regulations as a requirement, then licensees and applicants must comply with that standard as set forth in the regulation. If the secondary reference has been endorsed in an 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 an 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. Quality Group A

Quality Group A, while not explicitly defined in 10 CFR 50.55a(c), corresponds to the category of components presented in 10 CFR 50.55a(c)(1). As stated in 10 CFR 50.55a(c)(1), these components must meet the requirements for Class 1 components in Section III of the ASME BPV Code. This category is limited to components that are part of the reactor coolant pressure boundary, except for the portions excluded by 10 CFR 50.55a(c)(2) that are included in Quality Group B below. This exclusion applies to components whose failure would not prevent the reactor from being shut down and cooled down in an orderly fashion with normal makeup and components that are or can be isolated from the reactor coolant system by two valves in series (with automatic closure of open valves). While 10 CFR 50.55a does not explicitly refer to Quality Group A, it has been defined as such since the initial revision of this RG. Because this scope is well defined, this RG presents no further guidance for Quality Group A.

2. Quality Group B

The Quality Group B standards given in Table 1 of this RG should be applied to water- and steam-containing pressure vessels, heat exchangers (other than turbines and condensers), storage tanks, piping, pumps, and valves that are either (1) part of the reactor coolant pressure boundary, as defined in 10 CFR 50.2, "Definitions," but excluded from the requirements of 10 CFR 50.55a(c)(1) for reactor coolant pressure boundary components pursuant to 10 CFR 50.55a(c)(2) (as mentioned above in the section on Quality Group A), or (2) not part of the reactor coolant pressure boundary but part of the following:

a. systems or portions of systems[3] important to safety that are designed for (1) emergency core cooling, (2) postaccident containment heat removal, or (3) postaccident fission product removal;

b. systems or portions of systems important to safety that are designed for (1) reactor shutdown or (2) residual heat removal;

c. those portions of the steam systems of boiling-water reactors extending from the outermost containment isolation valve up to but not including the turbine stop and bypass valves,[4] and connected piping up to and including the first valve that is either normally closed or capable of automatic closure during all modes of normal reactor operation; alternatively, for boiling-water reactors containing a shutoff valve (in addition to the two containment isolation valves) in the main steam line and the main feedwater line, those portions of the steam and feedwater systems extending from the outermost containment isolation valves up to and including the shutoff valve or the first valve that is either normally closed or capable of automatic closure during all modes of normal reactor operation;

3 The system boundary includes those portions of the system necessary to accomplish the specified safety function and connected piping up to and including the first valve (including a safety or relief valve) that is either normally closed or capable of automatic closure when the safety function is required. This footnote applies wherever the phrase "systems or portions of [those] systems" appears in this guide.

4 The turbine stop-valve and turbine bypass valve, although not included in Quality Group B, should be subjected to a quality assurance program at a level generally equivalent to Quality Group B.

d. those portions of the steam and feedwater systems of pressurized-water reactors extending from and including the secondary side of steam generators up to and including the outermost containment isolation valves, and connected piping up to and including the first valve (including a safety or relief valve) that is either normally closed or capable of automatic closure during all modes of normal reactor operation;

e. systems or portions of systems that are connected to the reactor coolant pressure boundary and are not capable of being isolated from the boundary during all modes of normal reactor operation by two valves, each of which is either normally closed or capable of automatic closure.

3. Quality Group C

The Quality Group C standards given in Table 1 of this RG should be applied to water-, steam-, and radioactive-waste-containing pressure vessels; heat exchangers (other than turbines and condensers); storage tanks; piping; pumps; and valves that are not part of the reactor coolant pressure boundary or included in Quality Group B but are part of the following:

a. cooling water and auxiliary feedwater systems or portions of those systems important to safety that are designed for (1) emergency core cooling, (2) postaccident containment heat removal, (3) postaccident containment atmosphere cleanup, or (4) residual heat removal from the reactor and from the spent fuel storage pool (including primary and secondary cooling systems), although Quality Group B includes portions of those systems that are required for their safety functions and that (1) do not operate during any mode of normal reactor operation and (2) cannot be tested adequately;

b. cooling water and seal water systems or portions of those systems important to safety that are designed for the functioning of components and systems important to safety, such as reactor coolant pumps, diesels, and the control room;

c. systems or portions of systems that are connected to the reactor coolant pressure boundary and are capable of being isolated from that boundary during all modes of normal reactor operation by two valves, each of which is either normally closed or capable of automatic closure[5];

d. systems, other than radioactive waste management systems,[6] not covered by Regulatory Positions 2.a through 2.c (above) that contain or may contain radioactive material and whose postulated failure would result in conservatively calculated potential offsite doses that exceed 0.1 rem total effective dose equivalent; only single component failures need be assumed for those systems located in Seismic Category I structures, and no credit should be taken for automatic isolation from other components in the system or for treatment of released material, unless the isolation or treatment capability is designed to the appropriate seismic and quality group standards and can withstand loss of offsite power and a single failure of an active component.

4. Quality Group D

The Quality Group D standards given in Table 1 of this RG should be applied to water- and steam-containing components that are not part of the reactor coolant pressure boundary or included in

5 Components in influent lines may be classified as Quality Group D if they are capable of being isolated from the reactor coolant pressure boundary by an additional valve that has high leaktight integrity.

6 As noted above, RG 1.143 provides specific guidance on the classification of radioactive waste management systems.

Quality Groups B or C but that are part of systems or portions of systems that contain or may contain radioactive material.

Table 1: Quality Standards of LWR Quality Groups B, C, and D[7]

Components	QUALITY STANDARDS		
	Quality Group B	**Quality Group C**	**Quality Group D**
Pressure Vessels	ASME BPV Code, Section III, "Rules for Construction of Nuclear Facility Components,"[8] Class 2	ASME BPV Code, Section III, "Rules for Construction of Nuclear Facility Components,"[8] Class 3	ASME BPV Code, Section VIII, Division 1, "Rules for Construction of Pressure Vessels" (Ref. 15)
Piping	Class 2	Class 3	ASME B31.1, "Power Piping" (Ref. 16)
Pumps	Class 2	Class 3	Manufacturers' standards
Valves	Class 2	Class 3	ASME B31.1
Atmospheric Storage Tanks	Class 2	Class 3	API-650, "Welded Steel Tanks for Oil Storage" (Ref. 17); AWWA D-100, "Welded Steel Tanks for Water Storage" (Ref. 18); or ASME B96.1, "Welded Aluminum-Alloy Storage Tanks" (Ref. 19)
Storage Tanks (0-15 pounds per square inch gauge)	Class 2	Class 3	API-620, "Design and Construction of Large, Welded, Low-Pressure Storage Tanks" (Ref. 20)

7 Table 1 might be implemented in a different manner for nuclear power plant licensees implementing 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors." In the *Federal Register* notice (69 FR 68008) for the 10 CFR 50.69 rule, the Commission stated that RISC-3 (safety-related low safety significant) and RISC-4 (nonsafety-related low safety significant) structures, systems, and components (SSCs) will be exempt from the special treatment requirements (STRs) for qualification methods for environmental conditions and effects and seismic conditions. Nevertheless, the Commission stated that RISC-3 SSCs continue to be required to be capable of performing their safety-related functions under applicable environmental conditions and effects and seismic conditions, albeit at a lower level of confidence as compared to RISC-1 (safety-related safety significant) SSCs. As specified by the Commission, a licensee implementing 10 CFR 50.69 must consider operating life (aging) and combinations of operating life parameters (synergistic effects) in the design of RISC-3 electrical equipment. The Commission noted that this is particularly important if the equipment contains materials which are known to be susceptible to significant degradation due to thermal, radiation, and/or wear (cyclic) aging including any known synergistic effects that could impair the ability of the equipment to meet its design-basis function. The Commission direction in the *Federal Register* notice for 10 CFR 50.69 regarding the capability of RISC-3 SSCs to be able to perform their safety functions under applicable environmental and seismic conditions is clear for licensees who have received a license amendment to implement a 10 CFR 50.69 program. With respect to both RISC-3 and RISC-4 SSCs, the Commission decided to remove the RISC-3 and RISC-4 SSCs from detailed, specific requirements that provide the high level of assurance. However, the Commission stated in the *Federal Register* notice that the functional requirements for these SSCs remain.

8 See 10 CFR 50.55a for guidance on the ASME BPV Code edition and addenda to be applied. In addition, other RGs or Commission regulations cover the specific applicability of code cases, where appropriate. Applicants and licensees proposing the use of code cases not covered by guides or regulations should demonstrate that an acceptable level of quality and safety would be achieved.

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. 21), 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” (Ref. 22). 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[9] 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.

9 In this section, “licensees” refers to licensees of nuclear power plants under 10 CFR Parts 50 and 52.

REFERENCES[10]

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

2. CFR, "Licenses, Certifications, and Approvals for Nuclear Power Plants," Part 52, Chapter 1, Title 10, "Energy."

3. American Society of Mechanical Engineers (ASME), *Boiler and Pressure Vessel Code*, Section III, "Rules for Construction of Nuclear Facility Components," New York, NY.[11]

4. ASME, *Operation and Maintenance of Nuclear Power Plants*, Division 1, "Section IST: Rules for Inservice Testing of Light-Water Reactor Power Plants," New York, NY.

5. U.S. Nuclear Regulatory Commission (NRC), NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Washington, DC.

6. NRC, Regulatory Guide (RG) 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants," Washington, DC.

7. NRC, RG 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance," Washington, DC.

8. NRC, "Codes and Standards for Nuclear Power Plants," *Federal Register*, Vol. 49, No. 52: p. 9711 (49 FR 9711), Washington, DC, March 15, 1984.[12]

9. NRC, Advisory Committee on Reactor Safeguards, "Review of RG 1.26, Revision 5, 'Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants,'" October 17, 2016, (ADAMS Accession No. ML16286A590).

10. NRC, Office of the Executive Director for Operations, "Review of Regulatory Guide 1.26, Revision 5, 'Quality Group Classifications and Standards for Water-, Steam, and Radioactive-Waste-Containing Components of Nuclear Power Plants,'" December 13, 2016. (ADAMS Accession No. ML16300A310).

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10 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 for free 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.

11 Copies of ASME standards may be purchased from ASME, Two Park Avenue, New York, New York 10016-5990; telephone (800) 843-2763. Purchase information is available through the ASME Web site store at http://www.asme.org/Codes/Publications/.

12 All *Federal Register* notices listed herein were issued by the NRC. Copies are available for inspection or copying for a fee from the NRC's PDR at 11555 Rockville Pike, Rockville, MD; the PDR's mailing address is USNRC PDR, Washington, DC 20555; telephone (301) 415-4737 or (800) 397-4209; e-mail pdr.resource@nrc.gov.

11. American National Standards Institute (ANSI) and American Nuclear Society (ANS), ANSI/ANS-58.14-2011, "Safety and Pressure Integrity Classification Criteria for Light Water Reactors," Washington, DC.[13]

12. Nuclear Energy Institute (NEI) 00-04, "10 CFR 50.69 SSC Categorization Guideline," Washington, DC, July 2005.[14]

13. NRC, Management Directive 6.6, "Regulatory Guides," Washington, DC, May 2, 2016, (ADAMS Accession No. ML18073A170).

14. International Atomic Energy Agency (IAEA) SSG-30, "Safety Classification of Structures, Systems and Components in Nuclear Power Plants," Vienna, Austria, 2014.[15]

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

16. ASME Standard B31.1, "Power Piping," New York, NY.

17. American Petroleum Institute (API), API-650, "Welded Steel Tanks for Oil Storage," Washington, DC.[16]

18. American Water Works Association, (AWWA) D-100, "Welded Steel Tanks for Water Storage," Denver, CO.[17]

19. ASME Standard B96.1, "Welded Aluminum-Alloy Storage Tanks," New York, NY.

20. API-620, "Design and Construction of Large, Welded, Low-Pressure Storage Tanks," Washington, DC.

21. NRC, Management Directive 8.4, "Management of Backfitting, Forward Fitting, Issue Finality, and Information Requests," Washington, DC, September 20, 2019, (ADAMS Accession No. ML18093B087).

22. CFR, "Licenses, Certifications, and Approvals for Nuclear Power Plants," Part 52, Chapter 1, Title 10, "Energy."

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13 Copies of ANSI standards may be purchased from ANSI, 1819 L Street, NW, Washington, DC 20036, on their Web site at http://webstore.ansi.org/; telephone (202) 293-8020; fax (202) 293-9287; or e-mail storemanager@ansi.org.

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

15 Copies of International Atomic Energy Agency (IAEA) documents may be obtained through their Web site: WWW.IAEA.Org/ or by writing the International Atomic Energy Agency, P.O. Box 100 Wagramer Strasse 5, A-1400 Vienna, Austria; telephone (+431) 2600-0; fax (+431) 2600-7; or e-mail Official.Mail@IAEA.Org.

16 Copies of API standards may be purchased through their Web site: http://global.ihs.com/?RID=API1&MID=Q023 or by contacting API Headquarters at 1220 L Street, NW Washington, DC 20005-4070; telephone (202) 682-8000; Web Site http://www.api.org/; or e-mail standards@api.org.

17 Copies of AWWA standards may be purchased from the American Water Works Association, 6666 W. Quincy Ave., Denver, CO 80235; telephone (202) 682-8000; Web site http://www.awwa.org/bookstore/Category.cfm?cat=ALLSTD.

APPENDIX A

ALTERNATIVE CLASSIFICATION FOR COMPONENTS IN LIGHT-WATER-COOLED NUCLEAR POWER PLANTS

A-1. Introduction

As an alternative to the method described in Regulatory Guide (RG) 1.26 for determining acceptable quality standards for Quality Group A, B, C, and D components, an applicant or licensee may propose other classification methods for components in light-water-reactor (LWR) nuclear power plants. The American Nuclear Society (ANS) has prepared American National Standards Institute (ANSI)/ANS-58.14-2011, "Safety and Pressure Integrity Classification Criteria for Light Water Reactors," which discusses one such classification method that an applicant or licensee may consider for use. An applicant or licensee may propose the classification method discussed in ANSI/ANS-58.14-2011 as an alternative means to comply with U.S. Nuclear Regulatory Commission (NRC) regulations, including General Design Criterion (GDC) 1, "Quality Standards and Records," of Appendix A, "General Design Criteria for Nuclear Power Plants," to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," and 10 CFR 50.55a, "Codes and Standards," subject to the considerations discussed in this appendix. Applicants and licensees remain responsible for satisfying the NRC regulations for structures, systems, and components (SSCs) important to safety at the applicable nuclear power plant.

A-2. Scope of ANSI/ANS-58.14-2011

ANSI/ANS-58.14-2011 is applicable to all SSCs and parts (including consumables) for LWR nuclear power plant designs. The standard assigns both safety classification and pressure integrity classification to SSCs and their parts. In that the scope of ANSI/ANS-58.14-2011 is broader than that of RG 1.26, the application of ANSI/ANS-58.14-2011 should be limited to the consideration of the pressure integrity classification of components containing water, steam, or radioactive material. For example, Section 6.1, "Pressure-retaining items," of ANSI/ANS-58.14-2011 relates pressure integrity classification to safety classification and is not applicable to the pressure integrity classification. Further, the appendices to ANSI/ANS-58.14-2011 are beyond the scope of RG 1.26 or provide examples that are not uniformly applicable due to plant-specific conditions.

ANSI/ANS-58.14-2011 should only be used for LWR nuclear power plants. The process of ANSI/ANS-58.14-2011 may be useful to designers to help form a holistic picture of the facility, its SSCs, and their relationships to perform the necessary functions to maintain safety. This process may also be useful in framing discussions on component classification with the NRC staff.

A-3. Interdependence of ANSI/ANS-58.14-2011

ANSI/ANS-58.14-2011 relies on the existing language of RG 1.26 through references to RG 1.26 as "applicable guidance" in footnotes. However, ANSI/ANS-58.14-2011 does not directly mention or address the radiological criteria of RG 1.26, including the radiation dose consequence classification requirements of RG 1.26 (see RG 1.26, Section C.3 and Section C.4, on Quality Group C and Quality Group D, respectively, for radiological criteria not addressed in ANSI/ANS-58.14-2011). Therefore, a user of ANSI/ANS-58.14-2011 should complement the application of ANSI/ANS-58.14-2011 with RG 1.26 guidance.

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A-4. Terminology Distinctions

Significant terminology differences exist between ANSI/ANS-58.14-2011 and RG 1.26 that should be resolved when applying ANSI/ANS-58.14-2011. For example, the definition of primary containment in ANSI/ANS-58.14-2011 does not conform to the definition found in NRC-endorsed guidance (such as ANS 56.2 (ANSI N271-1976), "Containment Isolation Provisions for Fluid Systems" (Ref. 1), endorsed by RG 1.141 (Ref. 2) of the same title) and the NRC regulations (such as Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," to 10 CFR Part 50). The NRC-endorsed guidance should be followed where an inconsistency is identified between RG 1.26 and ANSI/ANS-58.14-2011.

The NRC regulations in 10 CFR Part 50, Appendix A, are applicable to SSCs important to safety at a nuclear power plant. ANSI/ANS-58.14-2011 discusses SSCs in the context of safety-related functions. Therefore, a user of ANSI/ANS-58.14-2011 should consider the full scope of 10 CFR Part 50, Appendix A, when classifying plant components.

A-5. Class 1/Quality Group A

Section 5.1.1, "Class 1," of ANSI/ANS-58.14-2011 states that Class 1 components include pressure-retaining portions and supports of mechanical items that form part of the reactor coolant pressure boundary (RCPB) and whose failure could cause a loss of reactor coolant in excess of the reactor coolant normal makeup capability. The NRC regulations in 10 CFR 50.55a(c)(2) provide two exceptions for RCPB components: (1) those components whose failure does not prevent the reactor from being shut down and cooled down in an orderly manner, assuming makeup is provided by the reactor coolant makeup system, and (2) those components that can be isolated from the reactor coolant system (RCS) by two valves in series (with automatic closure of open valves). The user of ANSI/ANS-58.14-2011 should provide assurance that the regulatory requirements in 10 CFR 50.55a(c)(2) are met when applying ANSI/ANS-58.14-2011. In addition, the discussion of safety classification for components of the RCPB in Section 5.1.1 is not within the scope of plant component quality group classification.

A-6. Class 2/Quality Group B

The description of Class 2 components in Section 5.1.2, "Class 2," of ANSI/ANS-58.14-2011 is not consistent with the NRC-accepted definition of primary containment. The user of ANSI/ANS-58.14-2011 should ensure that the primary containment is classified as Class 2. The user should also specify whether Class 2 classification in ANSI/ANS-58.14-2011 will apply to those RCPB components that meet the exclusion criteria of 10 CFR 50.55a(c)(2). The user should also address the guidance for Quality Group B provided in RG 1.26 that describes secondary systems for pressurized-water reactor and boiling-water reactor nuclear power plants with respect to Class 2 components.

ANSI/ANS-58.14-2011 lists functions performed by components typically classified as Class 2. ANSI/ANS-58.14-2011 and RG 1.26 appear to be aligned with respect to the Class 2 functions. However, ANSI/ANS-58.14 prefixes some functions with the term "emergency," while RG 1.26 uses the term "postaccident" or, in the case of residual heat removal, uses no prefix. The user of ANSI/ANS-58.14-2011 should ensure that these terminology differences do not result in inconsistencies in the component classification with respect to GDC 1. For example, RG 1.26 specifies reactor shutdown, while ANSI/ANS-58.14-2011 specifies emergency negative reactivity insertion (scram). Also, RG 1.26 specifies systems or portions of systems that are connected to the RCPB and are not capable of being isolated from the boundary during all modes of normal reactor operation by two valves, each of which is either normally closed or capable of automatic closure, while ANSI/ANS-58.14-2011 refers to terms such as "not normally isolated" or "cannot be automatically isolated" or "are not isolated" following the design-basis event.

A-7. Class 3/Quality Group C

Section 5.1.3, "Class 3," in ANSI/ANS-58.14-2011 states, in part, that Class 3 applies to pressure-retaining portions and supports of items that are not assigned to Class 1 or Class 2 but are within the scope of codes and standards specified in available guidance and are relied upon to accomplish one or more safety-related functions. Among the Class 3 components, a user of ANSI/ANS-58.14-2011 should include those components that are important to safety but do not necessarily accomplish safety-related functions, such as primary and secondary residual heat removal systems for spent fuel storage pools. In accordance with the guidance in RG 1.26, components whose failure could result in a significant offsite release should also be included in Quality Group C.

A-8. Class 4/Quality Group D

Section 5.1.4, "Class 4," in ANSI/ANS-58.14-2011 states, in part, that Class 4 applies to pressure-retaining portions and supports of items that are not assigned to Classes 1, 2, or 3 but are within the scope of the codes and standards specified in available guidance and are subject to at least one significant licensing requirement or commitment. RG 1.26 indicates that any component that is part of a system that may contain radioactive material and is not included in Quality Group A, B, or C should be designated Quality Group D. A user of ANSI/ANS-58.14-2011 should prepare adequate justification for use of the definition provided in ANSI/ANS-58.14-2011 if proposing not to classify components that contain or may contain radioactive material as, at least, Quality Group D.

A-9. Treatment of Supports

ANSI/ANS-58.14-2011 indicates that a support that is within the scope of the American Society of Mechanical Engineers (ASME) *Boiler & Pressure Vessel Code* (BPV Code), Section III, "Rules for Construction of Nuclear Facility Components," or other codes and standards specifying pressure integrity criteria, are to be assigned to the same class as the item it supports. A user of ANSI/ANS-58.14-2011 should apply ASME BPV Code, Section III, Subsection NF, "Supports," based on the applicable Code of construction for ASME BPV Code, Section III, supports.

A-10. Application of Guidance

ANSI/ANS-58.14-2011 provides examples of functions credited to components in each pressure integrity class, but it does not provide guidance for discerning whether a particular application is typical of the examples. A user of ANSI/ANS-58.14-2011 should review the plant-specific design in comparison to the criteria in RG 1.26, because some topics addressed in RG 1.26, such as spent fuel pool cooling, are not addressed in ANSI/ANS-58.14-2011. The user should provide assurance that the regulatory requirements for Quality Group B and C components in 10 CFR 50.55a(d) and 10 CFR 50.55a(e) are implemented to meet the Class 2 and Class 3 requirements in Section III of the ASME BPV Code, respectively.

Section 5.2, "Interface criteria," in ANSI/ANS-58.14-2011 states that interface barriers or isolation devices are to be provided when failure of an item connected to a different item of another class could prevent the item of another class from performing its pressure-retaining function. This section of ANSI/ANS-58.14-2011 exceeds the scope of classification guidance by providing design guidelines. Interfaces that provide pressure boundary separation of classes should be classified as the more stringent class.

A-11. Containment Penetrations

The user of ANSI/ANS-58.14-2011 should classify fluid system penetrations through primary containment and the corresponding primary containment isolation devices (outside or inside containment) not less than Class 2. For system piping assigned a class less stringent than Class 2 on either side of the penetration that penetrates the primary containment boundary, the interface to Class 2 should be at its connection to a primary containment isolation valve or, for a closed loop, to a primary containment penetration assembly.

The guidance in Section 4.5.3.5.2, "Boundary criteria for fluid system lines penetrating primary containment," of ANSI/ANS-58.14-2011 for the boundary criteria for fluid system lines penetrating primary containment is inconsistent with regulatory requirements and guidance. The GDC in 10 CFR Part 50, Appendix A, provide specific valve configuration requirements for containment isolation. ANSI/ANS-58.14-2011 does not meet GDC 55, "Reactor Coolant Pressure Boundary Penetrating Containment," for systems connected to the RCS that also penetrate primary containment.

RG 1.26 includes regulatory positions for systems connected to the RCS that penetrate primary containment. Specific regulatory positions apply to system piping that extends beyond the outermost containment isolation valve. ANSI/ANS-58.14-2011 does not include guidance for some of those system piping arrangements.

ANSI/ANS-58.14-2011 contains design requirements and recommendations that exceed classification guidance. For example, the standard refers to isolation signals and use of check valves for specific applications. Where ANSI/ANS-58.14-2011 describes containment isolation provisions for fluid systems or instrument lines penetrating the primary reactor containment, the NRC staff considers the guidance provided in RG 1.141 and RG 1.11, "Instrument Lines Penetrating Primary Reactor Containment" (Ref. 3), applicable. The guidance in ANSI/ANS-58.14-2011 for isolation valves is not consistent with the NRC regulations and regulatory guidance. Therefore, the NRC does not accept the application of ANSI/ANS-58.14-2011 for containment isolation provisions for fluid systems or instrument lines penetrating the primary reactor containment where it does not satisfy the NRC regulatory requirements in 10 CFR Part 50, Appendix A, GDC 54 through 57, or the guidance in RG 1.141 or RG 1.11.

A-12. Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors

The NRC regulations in 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors," allow specific applicants and licensees to implement a process for risk-informed categorization of plant SSCs using probabilistic risk assessment and other means described in their procedures where consistent with the categorization process approved by the NRC. The categorization process considers active functions, passive pressure boundary functions, and functions relied upon to respond to initiating events to separate SSCs into risk-informed safety class (RISC)-1, RISC-2, RISC-3, and RISC-4 categories.

The NRC regulations in 10 CFR 50.69 define the RISC categories as follows:

- RISC-1 SSCs are safety-related SSCs that perform safety-significant functions.
- RISC-2 SSCs are nonsafety-related SSCs that perform safety-significant functions.
- RISC-3 SSCs are safety-related SSCs that perform low-safety-significant functions.
- RISC-4 SSCs are nonsafety-related SSCs that perform low-safety-significant functions.

The 10 CFR 50.69 regulations define a safety-significant function as a function whose degradation or loss could result in a significant adverse effect on defense in depth, safety margin, or risk.

Before an applicant or licensee is allowed to implement 10 CFR 50.69, the NRC must approve the risk-informed SSC categorization process. For example, a licensee must submit an application for a license amendment under 10 CFR 50.90, “Application for Amendment of License, Construction Permit, or Early Site Permit,” that contains the information required by 10 CFR 50.69(b)(2). The NRC will approve a licensee’s application of 10 CFR 50.69 by issuing a license amendment if it determines that the categorization process satisfies the requirements of 10 CFR 50.69(c).

The applicant or licensee should have implementing procedures for properly categorizing each component using 10 CFR 50.69. The plant procedures should be consistent with the NRC-approved categorization process as described in the applicable final safety analysis report and sufficiently detailed to provide assurance of proper categorization of components. The description of the categorization of SSCs into RISC-1, RISC-2, RISC-3, and RISC-4 categories should include the process to categorize the safety significance of components based on the active (mechanical and electrical) functions of a component, the passive functions of a component (pressure boundary), and, for those components that are modeled in the probabilistic risk assessment, the importance of the component to the risk estimates.

RG 1.201, “Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance” (Ref. 4), provides interim guidance for complying with the NRC’s requirements in 10 CFR 50.69, by using the process described in Revision 0 of Nuclear Energy Institute (NEI) 00-04, “10 CFR 50.69 SSC Categorization Guideline,” issued July 2005 (Ref. 5), to determine the safety significance of SSCs and place them into the appropriate RISC categories. The safety significance of SSCs is determined using an integrated decisionmaking process, which incorporates both risk and traditional engineering insights. The safety functions of SSCs include both the design-basis functions (derived from the safety-related definition) and functions credited for preventing or mitigating severe accidents. Treatment requirements are then commensurately applied to the categorized SSCs to maintain their functionality.

An applicant or licensee may apply the categorization process described in 10 CFR 50.69 as part of its safety classification of plant components if approved by NRC review. The applicant or licensee should establish classes or quality groups of plant components that comply with the NRC regulations subject to the considerations discussed in this appendix. The NRC staff will review the safety classification of the plant components combined with the categorization process allowed by 10 CFR 50.69.

A-13. Conclusion

The NRC staff considers the guidance in ANSI/ANS-58.14-2011 to provide acceptable criteria for the quality group classification of components in LWR nuclear power plants, and for the assignment of pressure integrity classes to pressure-retaining items, where applied consistent with the guidance in this appendix. Applicants and licensees continue to be responsible for satisfying the NRC regulations for SSCs important to safety at the applicable nuclear power plant. The provisions in 10 CFR 50.69 may be applied as part of the safety classification of LWR components as described in this appendix.

APPENDIX A REFERENCES

1. ANS 56.2 (ANSI N271-1976), "Containment Isolation Provisions for Fluid Systems," Washington, DC.

2. NRC, RG 1.141, "Containment Isolation Provisions for Fluid Systems," Washington, DC.

3. NRC, RG 1.11, "Instrument Lines Penetrating Primary Reactor Containment," Washington, DC.

4. NRC, RG 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance," Washington, DC.

5. Nuclear Energy Institute (NEI) 00-04, "10 CFR 50.69 SSC Categorization Guideline," Washington, DC, July 2005[1]

[1] Publications from NEI are available at their Web site: http://www.nei.org/ or by contacting the headquarters at Nuclear Energy Institute, 1776 I Street, NW, Washington, DC 20006-3708; telephone (202) 739-800; fax (202) 785-4019

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