U.S. NUCLEAR REGULATORY COMMISSION

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Proposed Revision 2 to Regulatory Guide 1.89

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ENVIRONMENTAL QUALIFICATION OF CERTAIN ELECTRIC EQUIPMENT IMPORTANT TO SAFETY FOR NUCLEAR POWER PLANTS

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 meet regulatory requirements for environmental qualification (EQ) of certain electric[1] equipment important to safety for nuclear power plants. It endorses, with clarifications, the "English" portion of the dual logo International Electrotechnical Commission (IEC)/Institute of Electrical and Electronic Engineers (IEEE) Standard (Std.) 60780-323, "Nuclear Facilities—Electrical Equipment Important to Safety—Qualification," Edition 1, 2016-02 (Ref. 1).

Applicability

This RG applies to 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). With respect to 10 CFR Part 50, this RG applies to holders of or an applicant for an operating license. With respect to 10 CFR Part 52, this RG applies to applicants and holders of combined licenses, standard design certifications, standard design approvals, and manufacturing licenses. This RG does not apply to nuclear power plants that have submitted certifications as required by 10 CFR 50.82(a)(1) and by 52.110(a).

Applicable Regulations

- 10 CFR Part 50 requires, among other things, that structures, systems, and components (SSCs) that are important to safety in a nuclear power plant must be designed to accommodate the effects of environmental conditions (i.e., remain functional under postulated design-basis events (DBEs)).

1 For this document, the terms "electric", "electronics" and "electrical" are considered synonymous.

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-1361. 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. ML20183A423. The regulatory analysis may be found in ADAMS under Accession No. ML20192A230.

- 10 CFR 50.49, “Environmental qualification of electric equipment important to safety for nuclear power plants,” requires that holders or applicants for an operating license issued under Part 50 shall establish a program for the EQ of electric equipment as defined in 10 CFR 50.49.

- 10 CFR 50.55a(h), “Protection and safety systems,” states that protection systems must meet the requirements of the IEEE Std. 603-1991, “Criteria for Safety Systems for Nuclear Power Generating Stations” (Ref. 4), or IEEE Std. 279-1971, “Criteria for Protection Systems for Nuclear Power Generating Stations” (Ref. 5), contingent on the date of construction permit issuance. The design basis criteria identified in those standards or, for plants with construction permits issued before January 1, 1971, the criteria identified in the licensing basis for such facilities, include the range of transient and steady state environmental conditions during normal, abnormal, and accident conditions during which the equipment must perform its safety functions.

- General Design Criterion (GDC) 4, “Environmental and dynamic effects design bases,” of Appendix A, “General Design Criteria for Nuclear Power Plants,” to 10 CFR Part 50, states, in part, that SSCs important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents.

- General requirements associated with equipment qualification appear in GDC 1, “Quality Standards and Records,” GDC 2, “Design Bases for Protection Against Natural Phenomena,” and GDC 23, “Protection System Failure Modes,” of Appendix A to 10 CFR Part 50.

- Criterion III, “Design Control,” Criterion XI, “Test Control,” and Criterion XVII, “Quality Assurance Records,” of Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants” to 10 CFR Part 50 require design control measures to verify the adequacy of the design, a test program to assure that all testing required to demonstrate that SSCs will perform satisfactorily in service is performed in accordance with written procedures, and the maintenance of sufficient records as evidence of quality assurance activities.

* 10 CFR Part 52 requires that SSCs important to safety in a nuclear power plant be designed to accommodate the effects of environmental conditions and that design control measures, such as testing, be used to check the adequacy of the design.

  - 10 CFR 50.49(a) requires that holders of a combined license or a manufacturing license issued under Part 52 shall establish a program for the EQ of electric equipment as defined in 10 CFR 50.49. For a manufacturing license as defined in 10 CFR 52.157, only electric equipment defined in 10 CFR 50.49(b) which is within the scope of the manufactured reactor must be included in the EQ program.

  - 10 CFR 52.47(a)(13) requires that an applicant for a certified design must provide the list of electrical equipment important to safety as specified in 10 CFR 50.49(d).

  - 10 CFR 52.79(a)(10) requires that an application for a combined license must provide a description of the program, and its implementation, of an EQ program for electrical equipment per 10 CFR 50.49(a). The applicant must also provide the list of electric equipment that is important to safety as defined by 10 CFR 50.49(d).

- o 10 CFR 52.137(a)(13) requires that an applicant for a standard design approval must provide the list of electric equipment that is important to safety as defined by 10 CFR 50.49(d).

Related Guidance

- NUREG-0800, Section 3.11, "Environmental Qualification of Mechanical and Electrical Equipment" (Ref. 6), identifies staff guidance for determining that all items of equipment that are important to safety (mechanical, electrical, and I&C equipment) are capable of performing their design safety functions under all normal environmental conditions, anticipated operational occurrences, and accident and post-accident environmental conditions. It includes all environmental conditions that may result from any normal mode of plant operation, anticipated operational occurrences, design-basis events (as defined in 10 CFR 50.49(b)(1)(ii)), post-design-basis events, and containment tests.

- Prior to the issuance of the EQ rule (10 CFR 50.49), the Commission (in Memorandum and Order CLI-80-21 dated May 23, 1980) directed the staff to use NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," Revision 1 (Ref. 7), and the Division of Operating Reactors (DOR) Guidelines, "Guidelines for Evaluating Qualification of Class 1E Electrical Equipment in Operating Reactors," November 3, 1979 (Ref. 8), as requirements that licensees and applicants must meet in order to satisfy the equipment qualification requirements of 10 CFR Part 50. At that time, NUREG-0588 consisted of what is now Part I of NUREG-0588 (i.e., only the "for comment version" of NUREG-0588).

- Upon its issuance, 10 CFR 50.49, which is based on Part I of NUREG-0588 (hereinafter "NUREG-0588") and the DOR Guidelines, did not require requalification of electric equipment by applicants for and holders of operating licenses for nuclear power plants previously required by NRC to qualify equipment in accordance with the DOR Guidelines or NUREG-0588 (Category I or II).

  According to NUREG-0588, all nuclear reactors with Operating Licenses as of May 23, 1980, will be evaluated by the staff against the DOR guidelines. In accordance with the statement of considerations for 10 CFR 50.49, Category I requirements of NUREG-0588, which supplement the recommendations of and apply to equipment qualified in accordance with IEEE Std. 323-1974, "IEEE Standard for Qualifying Class IE Equipment for Nuclear Power Generating Stations" (Ref. 9), apply to nuclear power plants for which the construction permit safety evaluation report was issued after July 1, 1974. Category II requirements, which supplement the recommendations of and apply to equipment qualified in accordance with IEEE Std. 323-1971, "IEEE Trial-Use Standard: General Guide for Qualifying Class I Electric Equipment for Nuclear Power Generating Stations" (Ref. 10), apply to nuclear power plants for which the construction permit safety evaluation report was issued prior to July 1, 1974. For plants whose Safety Evaluation Reports for construction permits were issued since July 1, 1974, the Commission has used RG 1.89.

- RG 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," describes a method that is acceptable to the staff of the NRC for use in complying with the agency's regulations with respect to satisfying criteria for accident monitoring instrumentation in nuclear power plants (Ref. 11).

- RG 1.164, "Dedication of Commercial-Grade Items for Use in Nuclear Power Plants," describes methods that the staff of the NRC considers acceptable in meeting regulatory requirements for dedication of commercial-grade items and services used in nuclear power plants (Ref. 12).

- RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" (Ref. 13), provides guidance to licensees of operating power reactors on acceptable applications of alternative source terms (ASTs); the scope, nature, and documentation of associated analyses and evaluations; consideration of impacts on analyzed risk; and content of submittals related to the use of ASTs in radiological consequence analyses at operating power reactors.

- RG 1.180, "Guidelines for Evaluating Electromagnetic and Radio Frequency Interference in Safety-Related Instrumentation and Control Systems" (Ref. 14), describes design, installation, and testing practices acceptable to the NRC staff for addressing the effects of EMI/RFI and power surges on safety-related I&C systems in a nuclear power plant environment.

Recognizing that this RG and IEC/IEEE 60780-323, Edition 1, 2016-02, provide the fundamental approach for establishing EQ of electrical equipment in general, the following regulatory guidance documents include additional information for qualifying specific equipment or provide an additional level of detail for qualifying equipment:

- RG 1.209, "Guidelines for Environmental Qualification of Safety-Related Computer-Based Instrumentation and Control Systems in Nuclear Power Plants," describes a method that the NRC staff considers acceptable for determining the EQ procedures for safety-related computer-based I&C systems for service within nuclear power plants (Ref. 15).

- RG 1.40, "Qualification of Continuous Duty Safety-Related Motors for Nuclear Power Plants," describes a method that the staff of the NRC considers acceptable to implement regulatory requirements with regard to the design, inspection, and testing of normal atmosphere cleanup systems for controlling releases of airborne radioactive materials to the environment during normal operations, including anticipated operational occurrences (Ref. 16).

- RG 1.63, "Electric Penetration Assemblies in Containment Structures for Nuclear Power Plants," describes a method acceptable to the NRC staff for complying with the Commission's regulations for the design, construction, testing, qualification, installation, and external circuit protection of electric penetration assemblies in containment structures of nuclear power plants (Ref. 17).

- RG 1.73, "Qualification Tests for Safety-Related Actuators in Nuclear Power Plants," describes methods that the staff considers acceptable for the environmental qualification of safety-related power-operated valve actuators in nuclear power plants (Ref. 18).

- RG 1.100, "Seismic Qualification of Electric and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants," describes methods that the staff of the NRC considers acceptable for use in the seismic qualification of electrical and active mechanical equipment and the functional qualification of active mechanical equipment for nuclear power plants (Ref. 19).

- RG 1.153, "Criteria for Safety Systems," describes a method acceptable to the NRC staff for complying with the Commission's regulations with respect to the design, reliability, qualification, and testability of the power, instrumentation, and control portions of safety systems of nuclear plants (Ref. 20).

- RG 1.156, "Qualification of Connection Assemblies for Nuclear Power Plants," describes a method that the staff of the NRC considers acceptable for complying with the Commission's regulations on the EQ of connection assemblies and environmental seals in combination with cables or wires as assemblies for service in nuclear power plants. The EQ helps ensure that connection assemblies can perform their safety functions during and after a design-basis event (Ref. 21).

- RG 1.158, "Qualification of Safety-Related Lead Storage Batteries for Nuclear Power Plants," describes methods and procedures the staff of the NRC considers acceptable for use in complying with NRC regulations regarding the qualification method of safety-related lead-acid storage batteries for nuclear power plants (Ref. 22).

- RG 1.210, "Qualification of Safety-Related Battery Chargers and Inverters for Nuclear Power Plants," describes a method that the staff of the NRC considers acceptable in complying with the agency's regulations for the qualification of safety-related battery chargers and inverters for nuclear power plants (Ref. 23).

- RG 1.211, "Qualification of Safety-Related Cables and Field Splices for Nuclear Power Plants," describes a method that the staff of the NRC considers acceptable for complying with the Commission's regulations for the qualification of safety-related cables and field splices for nuclear power plants (Ref. 24).

- RG 1.213, "Qualification of Safety-Related Motor Control Centers for Nuclear Power Plants," describes a method that the staff of the NRC deems acceptable for complying with the Commission's regulations for qualification of safety-related motor control centers for nuclear power plants (Ref. 25).

Purpose of Regulatory Guides

The NRC issues RGs to describe methods that the staff considers acceptable for use in implementing specific parts of the agency's regulations, to explain techniques that the staff uses in evaluating specific problems or postulated events, and to provide guidance to applicants. Regulatory guides are not substitutes for regulations and compliance with them is not required. Methods and solutions that differ from those set forth in RGs will be deemed acceptable if they provide a basis for the findings required 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 Information Services 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

The previous revision of RG 1.89 was issued in June 1984 and endorsed the use of IEEE Std. 323-1974. The IEEE updated this standard in 1983 and 2003. However, the NRC did not officially endorse these versions in a regulatory guidance document.

In 2016, the IEEE standard was issued as a joint logo International Standard with IEC (IEC/IEEE 60780-323, Edition 1, 2016-02). The joint standard describes principles, methods, and procedures for qualifying, maintaining, and extending qualification, as well as updating qualification, of safety-related electrical equipment that is important to safety and interfaces that are to be used in nuclear power plants, including components or equipment of any interface whose failure could adversely affect any safety-related equipment.

The staff is revising RG 1.89 to endorse IEC/IEEE 60780-323, Edition 1, 2016-02, with clarifications, as this standard reflects almost 40 years of experience gained in implementing regulatory requirements and industry research and testing related to EQ. Nuclear plant license renewal provides additional motivation for continuing attention to equipment qualification. This revised guide contains information specific for EQ for both older plants and newer reactors licensed under both 10 CFR Parts 50 and 52.

Background

It is essential that safety-related electric equipment that is important to safety be qualified to demonstrate that it can perform its safety function under the environmental service conditions in which it will be required to function and for the length of time its function is required. Nonsafety-related electric equipment covered by 10 CFR 50.49(b)(2) must also be able to withstand environmental stresses caused by design-basis events under which its failure could prevent the satisfactory accomplishment of safety functions. This concept applies throughout this guide. The specific environment for which individual electric equipment must be qualified will depend on the installed location and the conditions under which it is required to perform its safety function. The requirements on electrical equipment located in a harsh environment are more stringent, since these components are generally not serviceable (i.e., not able to be accessed to replace or maintain) after the onset of a design basis event.

For the purposes of this guide, the primary objective of "qualification" is to demonstrate that equipment important to safety can perform its safety function(s) without experiencing common-cause failures before, during, and after applicable design-basis events. Chapter 11 and Appendix A to the Electric Power Research Institute's (EPRI's) "Plant Support Engineering: Nuclear Power Plant Equipment Qualification Reference Manual," Revision 1, issued September 2010 (Ref. 26), provides a detailed regulatory history of electrical and mechanical equipment qualification. While the agency has not officially endorsed this EPRI document, the NRC staff has reviewed Chapter 11 and Appendix A and found that it reflects an accurate representation of the regulatory history of electrical and mechanical equipment qualification.

IEC/IEEE 60780-323, Edition 1, 2016-02, was prepared by Subcommittee 45A, "Instrumentation, control and electrical systems of nuclear facilities," of IEC Technical Committee 45, "Nuclear instrumentation," in cooperation with the Nuclear Power Engineering Committee of the Power & Energy Society of the IEEE, under the IEC/IEEE Dual Logo Agreement between the IEC and the IEEE. The international standard describes principles, methods, and procedures for qualifying equipment, maintaining and extending qualification, and updating qualification for Class 1E equipment and interfaces that are to be used in nuclear power plants, including components or equipment of any interface whose failure could adversely affect any Class 1E equipment. The qualification specifications in IEC/IEEE 60780-323, Edition 1, 2016-02, when met, demonstrate and document the ability of equipment to perform safety function(s) under applicable service conditions, including design-basis events, reducing the risk of common-cause equipment failure.

The regulatory positions delineated in this guide reflect the state of the art. NRC and industry research programs currently in progress are investigating such concerns as the effects of oxygen in a loss of coolant accident (LOCA) environment, the validity of sequential versus simultaneous applications of steam and radiation environments, and fission product releases following accidents. The staff recognizes that the results of research programs may lead to revisions of the regulatory positions contained herein.

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 Standards 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 to the Commission's International Policy Statement (Ref. 27) and Management Directive and Handbook 6.6 (Ref. 28). In development of this RG, the NRC considered IAEA Safety Report Series No. 3, "Equipment Qualification in Operational Nuclear Power Plants: Upgrading, Preserving, and Reviewing," issued April 1998 (Ref. 29).

Furthermore, IEC/IEEE Std. 60780-323, Edition 1, 2016-02, was created based on a collaborative international effort to harmonize standard qualification practices developed from IEC 60780:1998, "Nuclear Power Plants – Electrical Equipment of the Safety System – Qualification" (Ref. 30), and IEEE Std. 323-2003, "IEEE Standard for Qualifying Class 1E Electrical Equipment for Nuclear Power Generating Stations" (Ref. 31).

Documents Discussed in Staff Regulatory Guidance

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

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.

the secondary reference, if appropriately justified, consistent with current regulatory practice, and consistent with applicable NRC requirements.

C. STAFF REGULATORY GUIDANCE

1. Staff Position 1 represents the NRC's endorsement of IEC/IEEE Std. 60780-323, Edition 1, 2016-02 with the following clarifications:

   a. 10 CFR 50.49(e)(5) requires, in part, that equipment qualified by test must be preconditioned by natural or artificial (accelerated) aging to its end-of-installed-life condition. Therefore, "end condition," as defined in Section 3.10 of IEC/IEEE Std. 60780-323, Edition 1, 2016-02, should be considered equivalent to "end-of-installed life." Note: Qualified equipment must be capable of performing its design function at the end-of-installed life.

   b. The following description and definition of "important to safety" should be used instead of the definition in Section 3.12 of IEC/IEEE Std. 60780-323, Edition 1, 2016-02:

      10 CFR 50.49 requires safety-related electric equipment (Class 1E) as defined in 10 CFR 50.49(b)(l) to be qualified to perform its intended safety functions. Appendix A to this guide lists typical safety-related equipment and systems. 10 CFR 50.49(b)(2) requires that nonsafety-related electric equipment be environmentally qualified if its failure under postulated environmental conditions could prevent satisfactory accomplishment of the safety functions specified in 10 CFR 50.49(b)(1)(i)(A) through (C) by safety-related electric equipment. Appendix B to this guide includes typical examples of nonsafety-related electric equipment. 10 CFR 50.49(b)(3) requires that certain post-accident monitoring equipment also be environmentally qualified. RG 1.97 includes regulatory guidance for post-accident monitoring equipment.

      While the above describes the electric equipment that is within the scope of 10 CFR 50.49, the NRC, in the introduction to 10 CFR Part 50, Appendix A, also states that "important to safety" SSCs are those SSCs that provide reasonable assurance that the facility can be operated without undue risk to public health and safety.

   c. The following definition of "qualified life" should be used instead of the definition in Section 3.20 of IEC/IEEE Std. 60780-323, Edition 1, 2016-02: "period for which an equipment has been demonstrated, through testing, analysis and/or experience, to be capable of remaining functional during and following design basis events to ensure that the criteria specified in 10 CFR 50.49(b)(1)(i)(A), (B), and (C) are satisfied."

      10 CFR 50.49(b)(1)(i) addresses the EQ functional requirements and states as follows:

      > This equipment is that relied upon to remain functional during and following design basis events to ensure—
      > (A) The integrity of the reactor coolant pressure boundary;
      > (B) The capability to shut down the reactor and maintain it in a safe shutdown condition; or
      > (C) The capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guidelines in 10 CFR 50.34(a)(1), 50.67(b)(2), or 100.11 of this chapter, as applicable.

   d. The term "service life," as defined in Section 3.22 of IEC/IEEE Std. 60780-323, Edition 1, 2016-02, implies that aging effects are insignificant unless the equipment is in service. However, the period before the operational phase of the SSC (i.e., shelf life) could also adversely impact the qualified life.

Therefore, the following definition of "service life" should be used instead of the definition in Section 3.22 of IEC/IEEE Std. 60780-323, Edition 1, 2016-02: "period from initial operation to final withdrawal from service of a structure, system, or component."

Note: The period before the operational phase of the structure, system, or component (i.e., shelf life), could also adversely impact the qualified life and, therefore, should be addressed.

e. Paragraph 4 of Section 5.1 of IEC/IEEE Std. 60780-323, Edition 1, 2016-02, notes that "Requirements, including EMC [Electromagnetic Compatibility], environmental/operational ageing and seismic requirements shall be specified in the design/purchase specifications." The prerequisite for aging electric equipment located in a mild environment is not within the scope of 10 CFR 50.49. Therefore, this sentence should be amended as follows: "Requirements, including EMC and seismic requirements, shall be specified in the design/purchase specifications."

f. Condition monitoring and associated condition-based qualification methodologies discussed in Section 6.3 of IEC/IEEE Std. 60780-323, Edition 1, 2016-02, represent new approaches for extending or establishing the qualified life of electrical equipment. If used, these methodologies must ensure that equipment important to safety will perform under the conditions specified in 10 CFR 50.49.

g. Section 7.2.6 of IEC/IEEE Std. 60780-323, Edition 1, 2016-02, should be supplemented with the following:

   Temperature and Pressure Conditions Inside Containment for a LOCA and Main Steam Line Break (MSLB). The following methods are acceptable to the NRC staff for calculating and establishing the containment pressure and temperature envelopes to which equipment should be qualified:

   (1) Methods for calculating mass and energy release rates for LOCAs and MSLBs are referenced in Appendix C to this guide. The calculations should account for the time dependence and spatial distribution of these variables. For example, superheated steam followed by saturated steam may be a limiting condition and should be considered.

   (2) For pressurized water reactors (PWRs) with a dry containment, LOCA or MSLB containment environment should be calculated using CONTEMPT-LT or equivalent industry codes.

   (3) For PWRs with an ice condenser containment, LOCA or MSLB containment environment should be calculated using LOTIC or equivalent industry codes.

   (4) For boiling water reactors (BWRs) with a Mark I, II, or III containment, LOCA or MSLB environment should be calculated using CONTEMPT-LT or equivalent industry codes.

h. Section 7.2.6.1 of IEC/IEEE Std. 60780-323, Edition 1, 2016-02, should be supplemented with the following:

   The radiation environment for qualification of electric equipment should be based on the radiation environment normally expected over the installed life of the equipment plus that associated with the most severe design basis accident during or following which the equipment must remain functional. The accident-related environmental conditions should be assumed to occur at the end of the installed life of the equipment. Methods acceptable to the NRC staff for establishing radiation doses for the qualification of equipment for BWRs and PWRs are provided in Appendix D and the following:

(1) Electric equipment that could be exposed to radiation should be environmentally qualified to a radiation dose that simulates the calculated radiation environment (normal and accident) that the equipment should withstand prior to completion of its required safety functions. Such qualification should consider that equipment damage is a function of total integrated dose and can be influenced by dose rate, energy spectrum, and particle type. The radiation qualification should factor in doses from all potential radiation sources at the equipment location. Plant-specific analysis should be used to justify any reductions in dose or dose rate resulting from component location or shielding. The qualification environment at the equipment location should be established using an analysis similar in nature and scope to that included in Appendix D to this guide and incorporating appropriate factors pertinent to the actual plant design (e.g., reactor type, containment design).

(2) Electric equipment that may be exposed to low-level radiation doses should not generally be considered exempt from radiation qualification testing. Exceptions may be based on qualification by analysis supported by test data or operating experience that verifies that the dose and dose rates will not degrade the operability of the equipment below acceptable values.

i. Section 7.2.6.4 of IEC/IEEE Std. 60780-323, Edition 1, 2016-02, should be supplemented with the following: "Electromagnetic conditions are generally independent of aging and design-basis events. Therefore, qualification can be established on a different sample than the sample subjected to aging and design-basis events."

j. Section 7.3.2 of IEC/IEEE Std. 60780-323, Edition 1, 2016-02, should be supplemented with the following:

(1) If synergistic effects have been known before the initiation of qualification, then the qualification program should account for them. The synergistic effect is the result of the combined environmental effects of the plant conditions such as radiation, humidity, and temperature that could result in greater degradation of equipment in relation to sequential application of the plant environment under normal, abnormal, and accident conditions. The synergistic effects on materials that are known to have such increased degradation under these conditions should be accounted for when assessing the qualified life.

(2) The expected operating temperature of the equipment under service conditions should be accounted for in thermal aging. The Arrhenius equation is considered an acceptable method of addressing accelerated thermal aging within the limitation of state-of-the-art technology. The use of other aging methods should be justified, and the staff will evaluate it on a case-by-case basis.

(3) The aging acceleration rate and activation energies used during qualification testing and the basis upon which the rate and activation energy were established should be defined, justified, and documented. Activation energy values should be based on the testing of the specific compound used in the equipment and on the most relevant material property and property endpoint (i.e., failure mechanism). Of note, the activation energy should be selected based on the temperature range of the equipment in service to ensure that the

equipment remains functional during and following a design-basis event. The selected activation energy values should be traceable to a specific test report for which these values were established, including the specific material property for which the activation energy was developed and how that material property is related to the function of the material in question. Potential nonlinearities and data extrapolation should be minimized by using activation energy values based on material test data obtained within the temperature range of interest. The data should also exhibit a good fit to the Arrhenius relationship within the applicable temperature range.

(4) Periodic surveillance and testing programs are acceptable to account for uncertainties about age-related degradation that could affect the functional capability of equipment. Results of such programs will be acceptable as ongoing qualification to modify the designated life (or qualified life) of equipment and should be incorporated into the maintenance and refurbishment/replacement schedules.

k. Section 7.4.1, "Type Testing," of IEC/IEEE Std. 60780-323, Edition 1, 2016-02, should be supplemented with the following:

(1) Electric equipment that could be submerged should be identified and qualified by testing in a submerged condition to demonstrate operability for the duration required. Analytical extrapolation of results for test periods shorter than the required duration should be justified.

(2) Electric equipment located in an area where rapid pressure changes are postulated simultaneously with the most adverse relative humidity should be qualified to demonstrate that the equipment seals and vapor barriers will prevent moisture from penetrating into the equipment to the degree necessary to maintain equipment functionality.

(3) The parameters to which electric equipment is being qualified (e.g., temperature, pressure, radiation) by exposure to a simulated environment in a test chamber should be measured sufficiently close to the equipment to ensure that actual test conditions accurately represent the environment characterized by the test.

(4) Performance characteristics that demonstrate the operability of equipment should be verified before, after, and periodically during testing throughout its range of required operability. Variables indicative of momentary failure that prevent the equipment from performing its safety function (e.g., momentary opening of a relay contact) should be monitored continuously to ensure that momentary failures (if any) have been accounted for during testing. For long-term testing, however, monitoring during periodic intervals may be used if justified.

(5) Chemical spray or demineralized water spray that is representative of service conditions should be incorporated during simulated event testing at pressure and temperature conditions that would occur when the spray systems actuate.

(6) Cobalt-60 or cesium-137 would be acceptable gamma radiation sources for EQ.

l. The suggested values in Section 7.4.1.7 of IEC/IEEE Std. 60780-323, Edition 1, 2016-02, are acceptable for meeting the requirements of 10 CFR 50.49(e)(8). Alternatively, quantified margins should be applied to the environmental parameters discussed in RG 1.183 to ensure that the postulated accident conditions have been enveloped during testing. These margins should be applied in addition to any conservatism used during the derivation of local environmental conditions of the equipment unless these conservatisms can be quantified and shown to contain appropriate margins. The margins should account for variations in commercial production of the equipment and the inaccuracies in the test equipment.

The design may require some electric equipment to perform its safety function only within the first 10 hours of the event. This equipment should remain functional in the accident environment for a period of at least 1 hour in excess of the time assumed in the accident analysis unless a time margin of less than 1 hour can be justified. This justification for each piece of equipment should include the following:

(1) consideration of a spectrum of breaks,

(2) the potential need for the equipment later in an event or during recovery operations,

(3) a determination that failure of the equipment after performance of its safety function will not be detrimental to plant safety or mislead the operator, and

(4) a determination that the margin applied to the minimum operability time, when combined with the other test margins, will account for the uncertainties associated with the use of analytical techniques in the derivation of environmental parameters, the number of units tested, production tolerances, and test equipment inaccuracies.

For all other equipment (e.g., post-accident monitoring), the 10-percent margin for equipment operating time identified in Section 7.4.1.7 should be used.

m. Section 7.4.1.9.3 of IEC/IEEE Std. 60780-323, Edition 1, 2016-02, should be supplemented with the following: "For insulating materials, a regression line (IEEE Std. 101, "IEEE Guide for the Statistical Analysis of Thermal Life Test Data" (Ref. 32)), may be used as a basis for selecting the aging time and temperature. Sample aging times of less than 100 hours should not be used."

n. Section 7.4.1.10 of IEC/IEEE Std. 60780-323, Edition 1, 2016-02, notes that a double peak test profile (with the same design-basis event temperature profile magnitude) is not required but may be used instead of one peak profile to increase the severity of the design-basis event test.

The concept of applying a double transient test profile first appeared in the 1974 version of IEEE Std. 323. Item 7 in Section 6.3.1.5 of IEEE Std. 323-1974 specified that the initial transient and the dwell at peak temperature needed to be applied at least twice. Appendix A to IEEE Std. 323-1974 also included a figure that showed a representative test chamber profile for a combined PWR and BWR test. IEEE Std. 323-1974 stated, in part, that if it is desired to qualify equipment for in-containment service for both PWRs and BWRs, the test conditions may be chosen to encompass both test profiles, including the chemical spray specified for PWRs and the temperature/pressure profile specified for BWRs. If the actual conditions are different from these curves, the parameters may be adjusted accordingly.

Section 7.4.1.10 should be supplemented to clarify the following:

(1) A double-transient should be used with equipment that may be vulnerable to thermal binding from different expansion rates of materials during the initial heatup.

(2) The use of double transients could help offset tests where the ramp rate (initial temperature rise) of the test is slower than the required profile. This is commonly the result of test chamber and steam supply limitations.

o. Section 8.3 of IEC/IEEE Std. 60780-323, Edition 1, 2016-02, discusses specific documentation requirements for equipment located in a mild environment. These documentation requirements are considered outside the scope of 10 CFR 50.49.

2. Staff Position 2 represents additional clarifications that were not addressed in IEC/IEEE Std. 60780-323, Edition 1, 2016-02:

a. In accordance with 10 CFR 50.49(l), replacement electric equipment installed after the issuance of 10 CFR 50.49 (February 22, 1983) must be qualified according to the provisions of 10 CFR 50.49 unless there are sound reasons to the contrary. The NRC staff considers the following to be sound reasons for the use of replacement equipment previously qualified in accordance with DOR Guidelines or NUREG-0588 in lieu of upgrading the qualification of electric equipment to the provisions of 10 CFR 50.49:

(1) The item of equipment to be replaced is a component of equipment that is routinely replaced as part of normal equipment maintenance (e.g., gaskets, O-rings, coils); these may be replaced with identical components.

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(2) The item to be replaced is a component that is part of an item of equipment qualified as an assembly; these may be replaced with identical components.

(3) Identical equipment to be used as a replacement was on hand as a part of the utility's stock before February 22, 1983. The shelf life of this equipment should be addressed for its potential impact on the qualified life.

(4) Replacement equipment qualified in accordance with 10 CFR 50.49 does not exist.

(5) Replacement equipment qualified in accordance with 10 CFR 50.49 is not available to meet installation and operation schedules. However, in such cases, the replacement equipment may be used only until upgraded equipment can be obtained and an outage of sufficient duration is available for replacement.

(6) Replacement equipment qualified in accordance with 10 CFR 50.49 would require significant plant modifications to accommodate its use.

(7) The use of replacement equipment qualified in accordance with 10 CFR 50.49 has a significant probability of creating human factor problems that would negatively affect plant safety and performance. Examples of this include the following:

(a) Knowledge, skills, and ability of existing plant staff would require significant upgrading to operate or maintain the specific replacement equipment;

(b) The use of the replacement equipment would create a one-of-a-kind application; or

(c) Maintenance, surveillance, or calibration activities would be unnecessarily complex.

b. When replacing previously qualified components, or subcomponents with commercially procured equipment, no significant changes in form, fit, or function should have occurred to the subcomponents since performance of the original qualification testing. This would include any changes to materials, material formulations, or critical manufacturing processes. Visual examinations or material-type verifications alone may not be sufficient to determine that relevant changes have not occurred. In such cases, a combination of material testing along with partial requalification testing of the components may be necessary, since an assessment of manufacturing process changes is typically not practical for commercially procured components. Additional guidance on maintaining qualification when procuring commercially procured replacement components appears in EPRI 3002002982, "Plant Engineering: Guideline for the Acceptance of Commercial-Grade Items in Nuclear Safety-Related Applications: Revision 1 to EPRI NP-5652 and TR-102260," published in September 2014 (Ref. 33), which the NRC endorsed in RG 1.164.

c. RG 1.209 states that metal oxide semiconductor devices generally have a lower radiation threshold than bipolar devices and are very sensitive to ionizing doses but relatively insensitive to neutron fluence. Therefore, radiation qualification for electronic components may have a lower exposure threshold. (As stated in Chapter 3, "Design of Structures, Components, Equipment, and Systems," of both NUREG-1503, "Final Safety Evaluation Report Related to the Certification of the Advanced Boiling Water Reactor Design," issued July 1994 (Ref. 34); and NUREG-1793, "Final Safety Evaluation Report Related to the Certification of the AP1000 Standard Design," issued September 2004 (Ref. 35), the staff considers a mild radiation environment for electronic equipment to be a total integrated dose less than 10 gray (Gy) (10^3 rad) and a mild radiation environment for other equipment to be less than 100 Gy (10^4 rad), to be acceptable.) An additional stressor to be considered in the qualification of digital systems is smoke exposure from an electrical fire. For smoke exposure, important failure mechanisms are not only long-term effects such as corrosion, but also short-term and perhaps intermittent malfunctions, such as leakage current. Smoke can cause circuit bridging and thus affect the operation of digital equipment. Because the edge connections and interfaces are typically uncoated, the most likely effect of the smoke is to impede communication and data transfer between subsystems. RG 1.209 provides several references that detail the effects of smoke exposure.

d. In 10 CFR 50.49(e)(5), the NRC calls for equipment qualified by test to be preconditioned by natural or artificial (accelerated) aging to its end-of-installed-life condition and further specifies that consideration must be given to all significant types of degradation that can affect the functional capability of the equipment. There are considerable uncertainties with regard to the processes and environmental factors that could result in such degradation. Diffusion-limited oxidation, synergisms, dose-rate effects, and inverse temperature are examples of such effects. Because of these uncertainties, state-of-the-art aging techniques are not capable of simulating all significant types of degradation, and natural pre-aging is not practical for producing timely results. As the state-of-the-art advances and uncertainties are resolved, techniques to simulate aging may become more effective. Experience suggests that consideration should be given, for example, to a combination of the following:

   (1) preconditioning of test samples employing the Arrhenius theory;

   (2) concurrent radiation and thermal aging or sequential aging, as well as the order of radiation and thermal aging, based on which produces the worst-case degradation; and,

   (3) surveillance, testing, and condition monitoring of selected equipment specifically directed toward detecting those degradation processes that are not amenable to

preconditioning and that could result in common-cause equipment failure during design basis accidents

e. Considerations such as the following should be taken into account when determining the environment for which the equipment is to be qualified: (1) equipment outside containment would generally see a less severe environment than equipment inside containment, (2) equipment whose location is shielded from a radiation source would generally receive a smaller radiation dose than equipment at the same distance from the source but exposed to direct radiation, (3) equipment required to initiate protective action would generally be required for a shorter period of time than instrumentation required to operate during and after an accident, and (4) analyses taking into account arrangements of equipment and radiation sources may be necessary to determine whether equipment needed for mitigation of design basis accidents other than LOCA or high-energy line breaks (HELB) could be exposed to a more severe environment than the plant-specific LOCA or HELB environments.

f. Electric equipment to be qualified in a nuclear radiation environment should be exposed to radiation, before testing, that simulates the calculated integrated dose (normal and accident) that the equipment must withstand before completion of its intended safety functions. Cobalt 60 or cesium 137 would be acceptable gamma radiation sources for EQ. As stated in 10 CFR 50.49(e)(4), EQ of safety related SSCs is required to address a radiation environment based on the “most severe design basis accident during or following which the equipment is required to remain functional.”

   In 10 CFR 100.11, “Determination of exclusion area, low population zone, and population center distance” (Ref. 36), the NRC provides criteria for evaluating the radiological aspects of the proposed site. A footnote to 10 CFR 100.11 states that the fission product release assumed in these evaluations should be based upon a major accident involving substantial meltdown of the core with subsequent release of appreciable quantities of fission products. The NRC cites Technical Information Document (TID) 14844, “Calculation of Distance Factors for Power and Test Reactor Sites” (Ref. 37), in 10 CFR Part 100, “Reactor Site Criteria,” as a source of further guidance on these analyses. Although initially used only for siting evaluations, the TID 14844 source term has been used for design-basis applications, such as EQ of equipment under 10 CFR 50.49. Regulations in 10 CFR 50.67, “Accident source term,” allows licensees to revise the accident source term used in design-basis radiological consequence analyses.

   RG 1.183 establishes an acceptable alternative source term (AST) and identifies the significant attributes of other ASTs that the NRC staff may find acceptable. For new reactor applications, the safety analysis requirements in 10 CFR 50.34(a)(1) and 10 CFR Part 52 (as applicable) include footnotes describing a fission product release similar to the one in the footnote to 10 CFR 100.11 described above. Although 10 CFR 50.49 does not include a similar footnote, power reactor license applicants have typically considered a core melt accident source term for the 10 CFR 50.49 EQ evaluation consistent with the footnote. Appendix D to this guide includes additional guidance on radiation EQ.

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. 38), 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.

REFERENCES [3]

1. International Electrotechnical Commission (IEC)/IEEE Std. 60780-323, "Nuclear Facilities — Electrical Equipment Important to Safety—Qualification," Edition 1, 2016-02.[4]

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

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

4. Institute of Electrical and Electronics Engineers (IEEE) Standard (Std.) 603-1991, "Criteria for Safety Systems for Nuclear Power Generating Stations," 1991, New York, NY.[5]

5. IEEE Std. 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations," 1971, New York, NY.

6. NRC, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Section 3.11, "Environmental Qualification of Mechanical and Electrical Equipment," Washington, DC.

7. NRC, NUREG-0588, Revision 1, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," Washington, DC, July 1981 (ADAMS Accession No. ML031480402).

8. NRC, Division of Operating Reactors Guidelines, "Guidelines for Evaluating Qualification of Class 1E Electrical Equipment in Operating Reactors," Washington, DC, November 3, 1979.

9. IEEE Std. 323-1971, "IEEE Trial-Use Standard: General Guide for Qualifying Class I Electric Equipment for Nuclear Power Generating Stations."

10. IEEE Std. 323-1974, "IEEE Standard for Qualifying Class IE Equipment for Nuclear Power Generating Stations."

11. NRC, RG 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," Washington, DC.

12. NRC, RG 1.164, "Dedication of Commercial-Grade Items for Use in Nuclear Power Plants," Washington, DC.

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3 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.

4 Copies may be obtained from the Institute of Electrical and Electronics Engineers, Inc., 3 Park Avenue, New York, NY 10016-5997.

5 Copies of Institute of Electrical and Electronics Engineers (IEEE) documents may be purchased from the Institute of Electrical and Electronics Engineers Service Center, 445 Hoes Lane, PO Box 1331, Piscataway, NJ 08855 or through the IEEE's public Web site at http://www.ieee.org/publications_standards/index.html.

13. NRC, RG 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” Washington, DC.

14. NRC, RG 1.180, “Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems,” Washington, DC.

15. NRC, RG 1.209, “Guidelines for Environmental Qualification of Safety-Related Computer-Based Instrumentation and Control Systems in Nuclear Power Plants,” Washington, DC.

16. NRC, RG 1.40, “Qualification of Continuous Duty Safety-Related Motors for Nuclear Power Plants,” Washington, DC.

17. NRC, RG 1.63, “Electric Penetration Assemblies in Containment Structures for Nuclear Power Plants,” Washington, DC.

18. NRC, RG 1.73, “Qualification Tests for Safety-Related Actuators in Nuclear Power Plants,” Washington, DC.

19. NRC, RG 1.100, “Seismic Qualification of Electric and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants,” Washington, DC.

20. NRC, RG 1.153, “Criteria for Safety Systems,” Washington, DC.

21. NRC, RG 1.156, “Qualification of Connection Assemblies for Nuclear Power Plants,” Washington, DC.

22. NRC, RG 1.158, “Qualification of Safety-Related Lead Storage Batteries for Nuclear Power Plants,” Washington, DC.

23. NRC, RG 1.210, “Qualification of Safety-Related Battery Chargers and Inverters for Nuclear Power Plants,” Washington, DC.

24. NRC, RG 1.211, “Qualification of Safety-Related Cables and Field Splices for Nuclear Power Plants,” Washington, DC.

25. NRC, RG 1.213, “Qualification of Safety-Related Motor Control Centers for Nuclear Power Plants,” Washington, DC.

26. Electric Power Research Institute, (EPRI) Nuclear Energy Institute (NEI) EPRI/NEI Report No. 1021067, “Plant Support Engineering: Nuclear Power Plant Equipment Qualification Reference Manual,” Palo Alto, CA, September 2010.[6]

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

28. NRC, Management Directive (MD) 6.6, “Regulatory Guides,” Washington, DC, May 2, 2016 (ADAMS Accession No. ML18073A170).

6 Copies of Electric Power Research Institute (EPRI) standards and reports may be purchased from EPRI, 3420 Hillview Ave., Palo Alto, CA 94304; telephone (800) 313-3774; fax (925) 609-1310.

29. IAEA Safety Reports Series No. 3, "Equipment Qualification in Operational Nuclear Power Plants: Upgrading, Preserving, and Reviewing," April 1998. [7]

30. IEC 60780:1998, "Nuclear Power Plants – Electrical Equipment of the Safety System – Qualification," October 1998.[8]

31. IEEE Std. 323-2003, "IEEE Standard for Qualifying Class 1E Electrical Equipment for Nuclear Power Generating Stations."

32. IEEE Std. 101-1987, "IEEE Guide for the Statistical Analysis of Thermal Life Test Data."

33. EPRI 3002002982, "Plant Engineering: Guideline for the Acceptance of Commercial-Grade Items in Nuclear Safety-Related Applications: Revision 1 to EPRI NP-5652 and TR-102260," published in September 2014 EPRI NP-5652," September 2014, Palto Alto, California.

34. NRC, NUREG 1503, "Final Safety Evaluation Report Related to the Certification of the Advanced Boiling Water Reactor Design," issued July 1994.

35. NRC, NUREG-1793, "Final Safety Evaluation Report Related to the Certification of the AP1000 Standard Design," issued September 2004, Initial Report.

36. CFR, "Reactor Site Criteria," Part 100, Chapter 1, Title 10, "Energy."

37. NRC, TID 14844, "Calculation of Distance Factors for Power and Test Reactor Sites," March 1962.

38. NRC, Management Directive 8.4, "Management of Backfitting, Forward Fitting, Issue Finality, and Information Requests."

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

8 Copies of International Electrical Commission (IEC) documents may be obtained through the IEC Web site: http://www.iec.ch/ or by writing the IEC Central Office at P.O. Box 131, 3 Rue de Varembé, 1211 Geneva, Switzerland, Telephone +41 22 919 02 11.

APPENDIX A

TYPICAL SAFETY-RELATED ELECTRIC EQUIPMENT OR SYSTEMS[9]

The following are typical safety-related electric equipment or systems:

- engineered safety features actuation
- reactor protection
- containment isolation
- steam line isolation
- main feedwater shutdown and isolation
- emergency power
- emergency core cooling
- containment heat removal
- containment fission product removal
- containment combustible gas control
- auxiliary feedwater
- containment ventilation
- containment radiation monitoring
- control room habitability system (e.g., heating, ventilation, and air conditioning; radiation filters)
- ventilation for areas containing safety equipment
- component cooling
- service water
- emergency systems to achieve safe shutdown

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9 In Title 10 of the *Code of Federal Regulations* (10 CFR) section 50.49(b)(1), the NRC identifies safety-related electric equipment as a subset of electric equipment important to safety and defines it as the equipment that is relied upon to remain functional during and following design-basis events to ensure (1) the integrity of the reactor coolant pressure boundary, (2) the capability to shut down the reactor and maintain it in a safe-shutdown condition, or (3) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guidelines in 10 CFR Part 100, "Reactor Site Criteria."

APPENDIX B

TYPICAL EXAMPLES OF NONSAFETY-RELATED EQUIPMENT

The equipment identified in Examples 1, 2, and 3 of this Appendix has typically been classified as safety related. However, the U.S. Nuclear Regulatory Commission licensed some operating plants using less definitive safety classification criteria, and these plants may contain nonsafety-related equipment, such as that in Examples 1, 2, and 3. The provisions of Title 10 of the *Code of Federal Regulations* (10 CFR) section 50.49, "Environmental qualification of electric equipment important to safety for nuclear power plants," require that the licensee provide appropriate EQ for equipment described in these examples, regardless of the safety classification of that equipment.

Example 4 applies to some plants, depending on the specific location of control system components.

Example 1

The injection of emergency feedwater (EFW) for pressurized-water reactors and the high-pressure coolant injection (HPCI) for boiling-water reactors are safety-related functions. The EFW system and the HPCI system are initiated upon detection of low water level. Automatic termination of these systems upon detection of high water level may also be provided. The high-level trip in some cases has been considered an equipment protection device; however, the inadvertent termination of EFW or HPCI due to misoperation of the level-sensing equipment when subjected to a harsh environment could defeat the safety-related injection function. Thus, the level sensing and electric equipment associated with automatic termination of the injection should be environmentally qualified.

Example 2

In some cases, the electrical control system for a pump (for example, a charging pump or an emergency core cooling system pump) will include termination commands on loss of lubrication oil pressure or low suction pressure. These features are provided for equipment protection. Failure of these features, however, would defeat the safety-related function. They should, therefore, be environmentally qualified.

Example 3

A safety-related fluid system may have nonsafety-related portions that are isolated from the safety-related portions upon the generation of an engineered safety features actuation signal. Isolation may be performed by motor-operated valves. These valve motor-operators should be environmentally qualified.

Example 4

Harsh environments associated with high-energy line breaks (HELBs) could cause control system malfunctions resulting in consequences more severe than those for the HELBs generally analyzed in Chapter 15 of each licensee's final safety analysis report or beyond the capability of operators or safety systems. In these cases, the control system failures could prevent satisfactory accomplishment of the safety functions required for the HELBs. The following are typical examples of control systems that could fail as a result of an HELB and whose consequential failure may not be bounded by HELBs analyzed in the final safety analysis report:

- automatic rod control system
- pressurizer power-operated relief valve control system
- main feedwater control system
- steam generator power-operated relief valve control system
- turbine generator control system

Based on the above, it may be necessary to environmentally qualify components associated with various control systems.

APPENDIX C[10]

METHODS FOR CALCULATING MASS AND ENERGY RELEASE

C-1. Loss-of-Coolant Accident

The following documents describe acceptable methods for calculating the mass and energy release to determine the loss-of-coolant accident environment for pressurized- and boiling-water reactors:

(1) Topical Report WCAP-8312A for Westinghouse plants is the non-proprietary version of WCAP-8264-P-A, "Westinghouse Mass and Energy Release Data for Containment Design," Revision 1, August 1975

(2) Section 6.2.1 of the preliminary safety analysis report for CESSAR System 80 for Combustion Engineering plants

(3) Appendix 6A to B-SAR-205 for Babcock & Wilcox plants

(4) NEDO-10320 and Supplements 1 and 2 for General Electric plants; NEDO-20533, dated June 1974, and Supplement 1, dated August 1975, for General Electric Mark III

C-2. Main Steamline Break

The following documents describe acceptable methods for calculating the mass and energy release to determine the main steamline break environment:

(1) Topical Report WCAP-8822 (MARVEL/TRANSFLA) for Westinghouse plants, noting that the use of this method is acceptable for all Westinghouse plants with the exception that a plant-specific containment temperature analysis will be required for ice condenser containments

(2) Appendix 6B to the preliminary safety analysis report for CESSAR System 80 for Combustion Engineering plants

(3) Section 15.1.14 of B-SAR-205 for Babcock & Wilcox plants

(4) NEDO 10320 and Supplements 1 and 2 for General Electric plants

10 Documents referenced in this Appendix are publicly available NRC published documents and 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.

APPENDIX D

QUALIFICATION IN THE RADIATION ENVIRONMENT

This appendix addresses assumptions associated with equipment qualification that are acceptable to the staff of the U.S. Nuclear Regulatory Commission (NRC) for performing radiological assessments for large light-water reactors similar to those currently in operation. The methodology and assumptions described within may not be appropriate for other reactor designs, and the use of the methodology and specific assumptions described in this section for other reactor designs must be demonstrated to be appropriate for the specific application and will be evaluated on a case by case basis. In addition, all applicants may propose alternative methods to address environmental qualification (EQ) requirements.

Regardless of the reactor design, consistent with Title 10 of the *Code of Federal Regulations* (10 CFR) section 50.49(e)(4), the methodology and assumptions for normal operation must be based on the total dose expected during normal operation over the installed life of the equipment, and the methodology and assumptions used for accident EQ must account for the radiation environment associated with the most severe design-basis accident for which the equipment is required to remain functional for the given reactor design. Therefore, the total integrated dose (i.e., the total of the normal operational and accident dose) for EQ must be based on the appropriate contribution of all significant radionuclides to which the equipment is exposed, considering the types of radiation (e.g., gamma, beta, and neutron radiation) and energy spectrum to which the equipment is exposed. As an alternative, a simplifying assumption may be used if the total integrated dose is justified to bound the total dose to the equipment and if sufficient margin is included to bound uncertainties associated with the assumptions.

D-1. Normal Operations Radiation Dose

The radiation environment resulting from normal operations should be based on the conservative source term estimates reported in the facility's safety analysis report or should be consistent with the primary coolant specific activity limits contained in the facility's technical specifications. The use of equilibrium primary coolant concentrations based on 1 percent of fuel cladding failures would be one acceptable method. In addition to sources resulting from fuel cladding failures, any additional sources of radiation exposure to the equipment should be considered if the source is of significance to the calculated normal operational dose. In estimating the integrated dose from prior normal operations, appropriate historical dose rate data may be used where available. The period of exposure for a normal operational dose is generally the duration of the plant license.

Neutron radiation should be appropriately considered for equipment near the core or that may be otherwise exposed to neutron radiation. Beta radiation normally need not be considered during normal operation, except for equipment in locations where beta radiation could be a significant contributor to the radiation dose (e.g., equipment that may be located inside a high gaseous radiation source). Shielded components need not be qualified to the beta radiation environment, provided that it can be demonstrated that the sensitive portions of the component or equipment are not exposed to significant beta radiation dose rates or that the effects of beta radiation, including heating and secondary radiation, have no deleterious effects on component performance.

The radiation exposure associated with maintenance, refueling, and anticipated operational occurrences (e.g., fuel transfer and resin sluicing) should be included in the normal operational dose for equipment exposed to additional radiation during or following these activities.

The amount of dose contributed by each of these sources depends on the location of the equipment, the time-dependent and location-dependent distribution of the source, and the effects of shielding.

D-2. Accident Radiation Dose

D-2.1 Basic Assumptions

Gamma and beta doses and dose rates should be determined for three types of radioactive source distributions: (1) activity suspended in the containment atmosphere, (2) activity plated out on containment surfaces, and (3) activity mixed in the containment sump water. A given piece of equipment may receive a dose contribution from any or all of these sources. The amount of dose contributed by each of these sources depends on the location of the equipment, the time-dependent and location-dependent distribution of the source, and the effects of shielding. For EQ components located outside the containment, additional radiation sources may include piping and components in systems that circulate containment sump water outside of containment. Activity deposited in ventilation and process filter media may be a source of post-accident dose. Shielded components need not be qualified to the beta radiation environment, provided that it can be demonstrated that the sensitive portions of the component or equipment are not exposed to significant beta radiation dose rates or that the effects of beta radiation, including heating and secondary radiation, have no deleterious effects on component performance.

The integrated dose should be determined from estimated dose rates using appropriate integration factors determined for each of the major source terms (e.g., containment sump, containment atmosphere, emergency core cooling system, normal operation). The period of exposure should be consistent with the survivability period for the EQ equipment being evaluated. The survivability period is the maximum duration, post-accident, that the particular EQ component is expected to operate and perform its intended safety function.

D-2.2 Fission Product Concentrations

The radioactivity released from the core during a design-basis loss-of-coolant accident should be based on the assumptions provided in Regulatory Position 3 and Appendix A to Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," or approved alternative assumptions. Although the design-basis LOCA is generally limiting for radiological EQ purposes, there may be certain components for which another design-basis accident may be limiting. In these cases, the assumptions in Appendices B through H to RG 1.183, as applicable, should be used, or approved alternative assumptions. EQ calculations may assume applicable features and mechanisms, provided that any prerequisites and limitations identified about their use are met. Additional considerations include the following:

- For pressurized-water reactor ice condenser containments, the source should be assumed to be initially released to the lower containment compartment. The distribution of the activity should be based on the forced recirculation fan flow rates and the transfer rates through the icebeds as functions of time.

- For boiling-water reactor Mark III designs, it should be assumed that all the activity initially is released to the drywell area and the transfer of activity from these regions by containment leakage to the surrounding reactor building volume should be used to predict the qualification levels within the reactor building (secondary containment).

D-2.3 Dose Model for Containment Atmosphere

The beta and gamma dose rates and integrated doses from the airborne activity within the containment atmosphere and from the plateout of aerosols on containment surfaces generally should be calculated for the midpoint in the containment, and this dose rate should be used for all exposed components. Radiation shielding afforded by internal structures may be neglected for modeling simplicity. More detailed calculations may be warranted for selected components if acceptable dose rates cannot be achieved using the simpler modeling assumptions.

Because of the short range of the betas in air, the airborne beta dose rates should be calculated using an infinite medium model. Other models, such as finite cloud and semi-infinite cloud, may be applicable to selected components with sufficient justification. The applicability of the semi-infinite model would depend on the location of the component, available shielding, and receptor geometry. For example, beta dose rates for equipment located on the walls of large containments or on large internal structures might be adequately assessed using the semi-infinite model. The staff will consider use of a finite cloud model on a case-by-case basis.

All gamma dose rates should be multiplied by a correction factor of 1.3 to account for the omission of the contribution from the decay chains of the radionuclides. This correction is particularly important for non-gamma-emitting radionuclides having gamma-emitting progeny; for example, cesium-137 decay to barium-137m. This correction may be omitted if the calculational method explicitly accounts for the emissions from buildup and decay of the radioactive progeny.

D-2.4 Dose Model for Containment Sumpwater Sources

With the exception of noble gases, all the activity released from the fuel should be assumed to be transported to the containment sump as it is released. This activity should be assumed to mix instantaneously and uniformly with other liquids that drain to the sump. This transport can also be modeled mechanistically as the time-dependent washout of airborne aerosols by the action of containment sprays. Radionuclides that do not become airborne on release from the reactor coolant system (e.g., they are entrained in non-flashed reactor coolant) should be assumed to be instantaneously transported to the sump and be uniformly distributed in the sump water.

The gamma and beta dose rates and the integrated doses should be calculated for a point located on the surface of the water at the centerline of the large pool of sump water. The effects of buildup should be considered. More detailed modeling with shielding analysis codes may be performed.

D-2.5 Dose Model for Equipment Located Outside Containment

EQ equipment located outside of containment may be exposed to (1) radiation from sources within the containment building, (2) radiation from activity contained in piping and components in systems that transport containment sump or reactor coolant system water outside of containment (e.g., emergency core cooling system, residual heat removal, sampling systems), (3) radiation from activity contained in piping and components in systems that may contain containment atmosphere (e.g., hydrogen recombiners, purge systems, sampling systems), (4) radiation from activity deposited in ventilation and process filter media, and (5) radiation from airborne activity in plant areas outside of the containment (i.e., leakage from recirculation systems). The amount of dose contributed by each of these sources is determined by the location of the equipment, the time-dependent and location-dependent distribution of the source, and the effects of shielding.

Because of the large amount of EQ equipment and the complexity of system and component layout in plant buildings, it is generally not reasonable to model each EQ component. A reasonable approach is to determine the limiting dose rate from all sources in a particular plant area (e.g., cubicle, floor, building) to a real or hypothetical receptor and to base the integrated doses for all components in that area on this postulated dose rate. Individual detailed modeling of selected equipment may be performed.

The integrated doses from components and piping in systems recirculating sump water should assume a source term based on the time-dependent containment sump source term described above. Similarly, the doses from components that contain air from the containment atmosphere should assume a source term based on the time-dependent containment atmosphere source term described above.

Analyses of integrated doses caused by radiation from the buildup of activity on ventilation and process filter media (e.g., filters, charcoal beds, resin beds) in systems containing containment sump water or atmosphere or both should assume that the ventilation or process flow is at its nominal design value and that the filter media is 100-percent efficient for iodine and particulates. The duration of flow through the filter media should be consistent with the plant design and operating procedures. Radioactive decay in the filter media should be considered. Shielding by structures and components between the filter and the EQ equipment may be considered.