U.S. NUCLEAR REGULATORY COMMISSION

REGULATORY GUIDE 1.89, REVISION 2

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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). This RG also provides guidance for addressing environmental stressors affecting the long-term reliability of electric equipment.

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 for a nuclear power plant. 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 52.110(a).

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

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Electronic copies of this RG, previous versions of RGs, and other recently issued guides are also available through the NRC's public web site in the NRC Library at https://www.nrc.gov/reading-rm/doc-collections/reg-guides/index.html/ under Document Collections, in Regulatory Guides. This RG is also available through the NRC's Agencywide Documents Access and Management System (ADAMS) at http://www.nrc.gov/reading-rm/adams.html, under ADAMS Accession Number (No.) ML22272A602. The regulatory analysis may be found in ADAMS under Accession No. ML20192A230. The associated draft guide DG-1361 may be found in ADAMS under Accession No. ML20183A423, and the staff responses to the public comments on DG-1361 may be found under ADAMS Accession No. ML22272A601.

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).

  - 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 for a nuclear power plant issued under Part 50 shall establish a program for the EQ of electric equipment as defined in 10 CFR 50.49. 10 CFR 50.49 also 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.

  - 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.

  - 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors," states in part that a holder of a license to operate a light-water reactor (LWR) nuclear power plant under 10 CFR Part 50; a holder of a renewed LWR license under 10 CFR Part 54; an applicant for a construction permit or operating license under 10 CFR Part 50; or an applicant for a design approval, a combined license, or manufacturing license under 10 CFR Part 52; may voluntarily comply with the requirements in 10 CFR 50.69 as an alternative to compliance with 10 CFR 50.49 for risk-informed safety class (RISC)-3 and RISC-4 SSCs.

    In the *Federal Register* (FR) notice (69 FR 68008) for the final rule establishing 10 CFR 50.69, the Commission stated that RISC-3 (safety-related low safety significant) and RISC-4 (non-safety-related low safety significant) SSCs will be exempt from the special treatment requirements 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 in the FR notice, 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 FR notice 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 FR notice that the functional requirements for these SSCs remain.

- 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.

- 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," requires, in part, that the pertinent requirements of this appendix apply to all activities affecting the safety-related functions of the structures, systems, and components that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public. These activities include designing, purchasing, fabricating, handling, shipping, storing, cleaning, erecting, installing, inspecting, testing, operating, maintaining, repairing, refueling, and modifying.

- 10 CFR Part 52 requires that SSCs important to safety in a nuclear power plant be designed to accommodate the effects of environmental conditions.

  - 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 required by 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 required by 10 CFR 50.49(d).

  - 10 CFR 52.97(b) requires that combined licenses must contain ITAAC that are necessary and sufficient to provide reasonable assurance that the facility has been constructed and will be operated in accordance with the license; the Atomic Energy Act of 1954, as amended; and NRC rules and regulations.

  - 10 CFR 52.99(c)(1) requires that each combined license holder notify the NRC that the prescribed inspections, tests, and analyses have been performed and that the prescribed acceptance criteria are met for each ITAAC included in their combined license.

  - 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 required by 10 CFR 50.49(d).

  - 10 CFR 52.157(f)(6) requires that an applicant for a manufacturing license provide a list of electric equipment important to safety that is required 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 instrumentation and control (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 1983 issuance of the 10 CFR 50.49 final rule, the Commission (in *Petition for Emergency and Remedial Action*, CLI-80-21, 11 NRC 707 (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 the 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, would be evaluated by the staff against the DOR Guidelines. As the Commission stated in the 10 CFR 50.49 final rule preamble, 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 NRC 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 regulatory requirements with respect to satisfying criteria for accident monitoring instrumentation in nuclear power plants (Ref. 11).

- RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" (Ref. 12), 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 1.183 can be used in radiological accident analysis and provides acceptable accident source term methodologies that may be used for EQ, as applicable. Therefore, for those applicants and licensees that RG 1.183 is applicable, RG 1.183 is referenced in this guide to describe acceptable source term methodologies to be used for EQ. However, RG 1.183 is not the only approved methodology for accident source terms and additional source term methodologies may be approved in the future. While other accident source term methodologies are not specifically referenced in this guide, approved accident source term methodologies for EQ may continue to be used (provided that they remain applicable) and new methodologies may be considered by the staff. The source term methodologies used must be applicable to the specific applicant or licensee and adequate to address EQ requirements.

- RG 1.215, "Guidance for ITAAC Closure Under 10 CFR Part 52," (Ref. 13) describes a method for documenting the completion of inspections, tests, analyses, and acceptance criteria.

The following documents facilitate qualification under other requirements, 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. 14). This regulatory guide addresses in part acceptable ways to address environmental qualification requirements covered within 10 CFR 50.49 and in part other environmental stressors (e.g., smoke and electromagnetic phenomena) that are not covered under 10 CFR 50.49.

- RG 1.40, "Qualification of Continuous Duty Safety-Related Motors for Nuclear Power Plants," endorses IEEE 334-2006, "Qualifying Continuous Duty Class 1E Motors for Nuclear Power Generating Stations," and describes a method that the staff of the NRC considers acceptable to implement regulatory requirements for the qualification of continuous duty safety-related motors for nuclear power plants. (Ref. 15).

- 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 regulatory requirements for the design, construction, testing, qualification, installation, and external circuit protection of electric penetration assemblies in containment structures of nuclear power plants (Ref. 16).

- 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. 17).

- 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. 18).

- RG 1.152, "Criteria for Use of Computers in Safety Systems of Nuclear Power Plants," (Ref. 19) endorses IEEE Std. 7-4.3.2-2003, "Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations," with exceptions and clarifications. IEEE Standard 7-4.3.2 contains criteria acceptable to the staff for addressing environmental design and qualification of computer-specific and programmable digital device-specific requirements and should be used in conjunction with NUREG-0588 and Regulatory Guide 1.89, as appropriate, for evaluating computer-specific requirements.

- RG 1.153, “Criteria for Safety Systems,” describes a method acceptable to the NRC staff for complying with the regulatory requirements for 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 regulatory requirements for 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 the regulatory requirements for the qualification method of safety-related lead-acid storage batteries for nuclear power plants (Ref. 22).

- RG 1.180, “Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems,” describes methods and procedures that the staff of the NRC considers acceptable for demonstrating compliance with the regulatory requirements for design, installation, and testing to address the effects of electromagnetic and radio-frequency interference (EMI/RFI), power surges, and electrostatic discharge on safety-related instrumentation and control (I&C) systems. (Ref. 23).

- 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 regulatory requirements for the qualification of safety-related battery chargers and inverters for nuclear power plants (Ref. 24).

- 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 regulatory requirements for the qualification of safety-related cables and field splices for nuclear power plants (Ref. 25).

- 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 regulatory requirements for qualification of safety-related motor control centers for nuclear power plants (Ref. 26).

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 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

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 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. Non-safety-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 placed on electrical equipment located in a harsh environment are more stringent because these components are generally not serviceable (i.e., not able to be accessed to be replaced or maintained) after the onset of a design-basis event. Requirements for dynamic and seismic qualification of electric equipment important to safety; protection of electric equipment important to safety against other natural phenomena and external events; and environmental qualification of electric equipment important to safety located in a mild environment are not included within the scope of 10 CFR 50.49.

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 during and after applicable design-basis events. 10 CFR 50.49(b)(1)(ii) defines "design-basis events" as conditions of normal operation, including anticipated operational occurrences, design-basis accidents, external events, and natural phenomena for which the plant must be designed to ensure the functions listed in 10 CFR 50.49(b)(1)(i)(A) through (C).

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 regarding the processes and environmental factors that could result in such degradation. Inverse temperature or reverse temperature effect is where polymer degradation occurs more rapidly for constant dose rates as the combined environment temperature is lowered. NUREG/CR 7153, “Expanded Materials Degradation Assessment (EMDA): Volume 5: Aging of Cables and Cable Systems,” Section 3.3, “Inverse Temperature,” provides additional information. Diffusion-limited oxidation, synergisms, dose rate effects, and inverse temperature effects are examples of such effects. Experience suggests that consideration should be given, for example, to 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 more severe 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.

When determining the environment for which the equipment is to be qualified, environmental 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 loss-of-coolant accidents (LOCA) or high-energy line breaks (HELB) could be exposed to a more severe environment than the plant-specific LOCA or HELB environments.

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 drywell 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 pressurized-water-reactor (PWR) and boiling-water reactor (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. As emphasized in IEC/IEEE Std. 60780 323, Edition 1, 2016 02, it is essential that the test profile envelopes the plant-specific design bases.

In SRM-SECY-05-0197, “Review of Operational Programs in a Combined License Application and General Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria,” the NRC staff describes operational programs for new nuclear power plants as programs that are required by regulation, are reviewed by the NRC staff for acceptability with the results documented in the safety evaluation report, and will be verified for implementation by NRC inspectors. For example, SECY-05-0197 specifies the EQ program as an operational program.

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. 27), provide 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.

Attachment 2, "Select Topics Regarding the Environmental Qualification Process," of Inspection Procedure 71111 Attachment 21N, "Design Bases Assurance Inspection (Programs)," (Ref. 28) provides additional clarification on select environmental qualification topics.

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[2] 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 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[3] pursuant to the Commission's International Policy Statement (Ref. 29) and Management Directive and Handbook 6.6 (Ref. 30). 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. 31).

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. 32), and IEEE Std. 323-2003, "IEEE Standard for Qualifying Class 1E Electrical Equipment for Nuclear Power Generating Stations" (Ref. 33).

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

2 As noted in 10 CFR 50.49, the NRC considers Class 1E to be synonymous with the term "safety-related."

3 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.

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

C. STAFF REGULATORY GUIDANCE

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

   a. Section 5.1, "Qualification objective," of IEC/IEEE Std. 60780-323, Edition 1, 2016-02, discusses the need for qualification of equipment to address environmental conditions. The purpose of this RG is to provide an acceptable approach for satisfying the environmental qualification requirements for certain electric equipment. The guidance endorsed in this RG could also be used to satisfy GDC 4 requirements for the design of structures, systems, and components important to safety to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including LOCAs.

   b. 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.

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

      The introduction to 10 CFR Part 50, Appendix A, 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.

      10 CFR 50.49 requires safety-related (Class 1E) electric equipment as defined in 10 CFR 50.49(b)(l) to be environmentally 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 non-safety-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 non-safety-related electric equipment that may be in the scope of 10 CFR 50.49. 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.

   d. 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) are satisfied."

   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." Guidance for demonstrating EMC and EMI/RFI qualification is provided in Regulatory Guide 1.180. While not within the scope of 10 CFR 50.49, the requirements for environmental design considerations of equipment located in a mild environment is covered by GDC 4 of Appendix A to 10 CFR Part 50. Guidance for demonstrating seismic qualification is provided in Regulatory Guide 1.100.

f. Reassessing qualified life discussed in Section 6.2 of IEC/IEEE Std. 60780-323, Edition 1, 2016-02, should be supplemented to include the following:

   X.E1, "Environmental Qualification of Electric Equipment," of NUREG-2191, "Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report," (Ref. 34) and X.E1, "Environmental Qualification (EQ) of Electric Components," of NUREG-1801, "Generic Aging Lessons Learned Report," (Ref. 35) note that under 10 CFR 54.21(c)(1)(iii), plant EQ programs, which implement the requirements of 10 CFR 50.49 (as further defined and clarified by the DOR Guidelines, NUREG-0588, and Regulatory Guide 1.89), are viewed as aging management programs for license renewal and subsequent license renewal. Reanalysis of an aging evaluation to extend the qualification of components under 10 CFR 50.49(e) is performed on a routine basis as part of an EQ program. Reanalysis evaluates the original attributes, assumptions and conservatisms for environmental conditions, and other factors of an aging evaluation to demonstrate that equipment qualified life can be extended. Important attributes for the reanalysis of an aging evaluation include analytical methods, data collection and reduction methods, underlying assumptions, acceptance criteria, and corrective actions. These attributes are discussed further in Section X.E1 of NUREG-2191 and NUREG-1801.

g. Condition monitoring and associated condition-based qualification methodologies discussed in Section 6.3 of IEC/IEEE Std. 60780-323, Edition 1, 2016-02, represent approaches for extending or establishing the qualified life of electrical equipment.

   Condition monitoring recognizes the fact that the aging process in a 10 CFR 50.49 test method qualification program can be an acceptable process of determining end of qualified life, if it is proven during a qualification by test program to be a condition indicator that must be measurable, change monotonically with time, be correlated with the safety function performance under design-basis event conditions, be linked to the functional degradation of the qualified equipment, and have a consistent trend from unaged through the limit of the qualified pre-accident condition.

h. 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). Containment response methodologies require review and approval by the NRC. 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) Typical 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 PWRs with a dry containment, LOCA or MSLB containment environment should be calculated using CONTEMPT-LT or equivalent industry codes.[4]

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4 Containment pressure and temperature environment should be calculated using codes that are consistent with the licensee's design and licensing basis.

(3) For PWRs with an ice condenser containment, LOCA or MSLB containment environment should be calculated using LOTIC or equivalent industry codes.[4]

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

i. 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 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 radiation sources that significantly impact the total integrated dose to the equipment. Facility-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 design of the facility (e.g., reactor type, containment design).

(2) Shielded components need be qualified only to the gamma radiation environment provided 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. If, after considering the appropriate shielding factors, the total beta radiation dose contribution to the equipment or component is calculated to be less than 10% of the total gamma radiation dose to which the equipment or component has been qualified, the equipment or component may be considered qualified for the beta and gamma radiation environment, provided that the total integrated dose to equipment remains conservative considering all assumptions made in the analysis, including margin.

j. Section 7.4.1.9.3 of IEC/IEEE Std. 60780-323, Edition 1, 2016-02, should be supplemented with the following:

(1) Synergistic effects must be considered when these effects are believed to have a significant effect on equipment performance. A 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 individual application of the plant environmental effects under normal, abnormal, and accident

conditions. If synergistic effects have been identified prior to the initiation of qualification, they should be accounted for in the qualification program. Synergistic effects known at this time are dose rate effects and effects resulting from the different sequence of applying radiation and (elevated) temperature.

(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. The selected activation energy should be representative of the most limiting material in a component/sub-component when determining qualified life. Activation energy values should be based on the testing of the material and address the most relevant/limiting material property(ies) and property endpoint(s) (i.e., failure mechanism(s)). 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 test report or other documentation 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.

IEEE Std. 98-2016, "IEEE Standard for the Preparation of Test Procedures for the Thermal Evaluation of Solid Electrical Insulating Materials," (Ref. 36) and IEEE Std. 99-2019, "IEEE Recommended Practice for the Preparation of Test Procedures for the Thermal Evaluation of Insulation Systems for Electrical Equipment," (Ref. 37) contains additional technical information and criteria useful for determining activation energy values. However, the staff is not officially endorsing these IEEE Standards in this RG.

(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, as applicable and depending on the equipment's safety function. 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 after the test chamber reaches the maximum 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). 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 organic materials, a regression line (IEEE Std. 101, 'IEEE Guide for the Statistical Analysis of Thermal Life Test Data' (Ref. 38)), 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. A regression line alone does not form a basis for the aging time and aging temperature. This approach provides an activation energy proportional to the slope of the regression line that can be used to determine the amount of time at the aging temperature to cause thermal aging equivalent to aging that would occur during the desired life at the expected temperature. The aging time and aging temperature are not a point on the regression line. Once the activation energy is determined, an aging time can be calculated for an assumed aging temperature or an aging temperature can be calculated for an assumed aging time."

n. Section 8 of IEC/IEEE Std. 60780-323, Edition 1, 2016-02, provides guidance on documentation. Additional documentation guidance can be found in Appendix E of this Regulatory Guide. 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.

      (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.

      (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. 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," (Ref. 39); and NUREG-1793, "Final Safety Evaluation Report Related to the Certification of the AP1000 Standard Design," (Ref. 40), 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.). While not a consideration under 10 CFR 50.49, an additional stressor that may need to be considered in the qualification of digital systems is smoke exposure from an electrical fire from operational conditions (e.g., 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.

c. Electric equipment to be qualified in a nuclear radiation environment should be exposed to radiation 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 required in 10 CFR 50.49(e)(4), the radiation environment must be based on the total dose expected during normal operations over the installed life of the equipment and the radiation environment associated with the most severe design-basis accident during or following which the equipment is required to remain functional. In addition, GDC 4 requires that SSCs important to safety shall be designed to accommodate the affects and to be compatible with the environmental conditions, including those associated with postulated accidents. RG 1.183 provides guidance on accident radiological source terms and may be used, as applicable, in combination with Appendix D to this guide for radiation equipment qualification. Alternative source terms and assumptions may be developed for assessing equipment qualification to the radiation environment. Any alternatives will be evaluated on a case-by-case basis.

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. 41), 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 [5]

1. International Electrotechnical Commission (IEC)/Institute of Electrical and Electronics Engineers (IEEE) Standard (Std.) 60780-323, "Nuclear Facilities —Electrical Equipment Important to Safety—Qualification," Edition 1, 2016-02.[6]

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

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

4. IEEE, Std. 603-1991, "Criteria for Safety Systems for Nuclear Power Generating Stations," 1991, New York, NY.[7]

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," Revision 3, Section 3.11, "Environmental Qualification of Mechanical and Electrical Equipment," Washington, DC, March 2007 (ML063600397).

7. NRC, NUREG-0588, Revision 1, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," Washington, DC, July 1981 (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. 323-1974, "IEEE Standard for Qualifying Class IE Equipment for Nuclear Power Generating Stations." 1974, New York, NY.

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

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

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

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5 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.

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

7 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 1.215, “Guidance for ITAAC Closure Under 10 CFR Part 52,” Washington, DC.

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

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

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

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

18. 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.

19. NRC, RG 1.152, “Criteria for Use of Computers in Safety Systems of 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 1.180, “Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems,” Washington, DC.

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

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

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

27. Electric Power Research Institute (EPRI), EPRI Report No. 1021067, “Plant Support Engineering: Nuclear Power Plant Equipment Qualification Reference Manual,” Palo Alto, CA, September 2010.[8]

28. NRC, IP 7111 Attachment 21N, “Design Bases Assurance Inspection (Programs),” Washington, DC, February 2019 (ML19036A556).

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

30. NRC, Management Directive (MD) 6.6, "Regulatory Guides," Washington, DC.

31. IAEA Safety Reports Series No. 3, "Equipment Qualification in Operational Nuclear Power Plants: Upgrading, Preserving, and Reviewing," April 1998. [9]

32. IEC 60780:1998, "Nuclear Power Plants – Electrical Equipment of the Safety System – Qualification," October 1998.[10]

33. IEEE Std. 3232003, "IEEE Standard for Qualifying Class 1E Electrical Equipment for Nuclear Power Generating Stations." 2003, New York, NY.

34. NRC, NUREG-2191, Volumes 1 and 2, "Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report," Washington, DC, July 2017 (ML17187A031 and ML17187A204).

35. NRC, NUREG-1801, "Generic Aging Lessons Learned Report," Washington, DC, July 2001, (ML012060392, ML012060514, ML012060539, and ML012060521).

36. IEEE Std. 98-2016, "IEEE Standard for the Preparation of Test Procedures for the Thermal Evaluation of Solid Electrical Insulating Materials," 2016, New York, NY.

37. IEEE Std. 99-2019, "IEEE Recommended Practice for the Preparation of Test Procedures for the Thermal Evaluation of Insulation Systems for Electrical Equipment," 2019, New York, NY.

38. IEEE Std. 101-1987, "IEEE Guide for the Statistical Analysis of Thermal Life Test Data," 1987, New York, NY.

39. NRC, NUREG-1503, "Final Safety Evaluation Report: Related to the Certification of the Advanced Boiling Water Reactor Design," Washington, DC.

40. NRC, NUREG-1793, "Final Safety Evaluation Report Related to the Certification of the AP1000 Standard Design," Washington, DC.

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

42. RG 1.75, "Physical Independence of Electric Systems," Washington, DC.

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9 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.

10 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[11]

For large light-water reactors, the following are typical safety-related electric equipment or systems. Some items on this list may not be applicable to passive designs, small modular designs, or advanced reactors. However, emergency systems to achieve safe shutdown could be safety-related electric systems for any design.

- 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
- 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

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11 Mechanical, electrical, and I&C equipment covered by Chapter 3.11, "Environmental Qualification of Mechanical and Electrical Equipment," of NUREG-0800, "Standard Review Plan," include the following: (A) Electrical equipment that are essential for shutting down the reactor and maintaining it in a safe shutdown condition, containment isolation, reactor core cooling, and containment and reactor heat removal, or otherwise are essential in preventing significant release of radioactive material to the environment following a design-basis accident; (B) Electrical equipment that initiates the above functions automatically; (C) Electrical equipment that is used by the operators to initiate the above functions manually; (D) Electrical equipment whose failure can prevent the satisfactory accomplishment of one or more of the above safety functions; (E) Other electrical equipment important to safety, as described in 10 CFR 50.49(b)(1) and (2); (F) Certain post-accident monitoring equipment, as described in 10 CFR 50.49(b)(3) and RG 1.97; and (G) Protection and safety systems as described in 10 CFR 50.55a(h) and RG 1.209.

APPENDIX B

TYPICAL EXAMPLES OF NON-SAFETY-RELATED EQUIPMENT

Associated circuits, as defined in Regulatory Guide 1.75, “Physical Independence of Electric Systems, (Ref. 42) need only be qualified to ensure that they will not fail under postulated environmental conditions in a manner that could prevent satisfactory accomplishment of safety functions by safety-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 non-safety-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 environmental qualification (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 mis-operation 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 non-safety-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[12]

TYPICAL METHODS FOR CALCULATING MASS AND ENERGY RELEASE[13]

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

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12 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/index 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.

13 Mass and energy releases are developed using a methodology that is consistent with the licensing basis of the plant. The listed methods are examples for existing designs.

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.

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 <u>Basic Assumptions</u>

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 design-basis event qualification for the EQ equipment being evaluated.

D-2.2 <u>Fission Product Concentrations</u>

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 the other appendices 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.

APPENDIX E

QUALIFICATION DOCUMENTATION FOR ELECTRIC EQUIPMENT

In order to ensure that an environmental qualification program conforms to General Design Criteria 1, 2, 4, and 23 of Appendix A; Sections III, XI, and XVII of Appendix B; and 10 CFR 50.49, the following information on the qualification program should be submitted to the NRC:

E-1 Provide a list of all electric equipment within the scope of this guide such as the following:

(1) Switchgear
(2) Motor control centers
(3) Valve operators and solenoid valves
(4) Motors
(5) Logic equipment
(6) Cable
(7) Connectors
(8) Sensors (pressure, pressure differential, temperature, flow and level, neutron, and other radiation)
(9) Limit switches
(10) Heaters
(11) Fans
(12) Control boards
(13) Instrument racks and panels
(14) Electric penetrations
(15) Splices
(16) Terminal blocks

E-2 For each item of equipment identified in item E-1, provide the following:

(1) Type (functional designation)
(2) Manufacturer
(3) Manufacturer's type number and model number
(4) Plant ID/tag number and location

E-3 Categorize the equipment identified in item E-1 into one of the following categories:

(1) Equipment that will experience the environmental conditions of design-basis accidents through which it must function to mitigate such accidents must be qualified to demonstrate operability in the accident environment for the time required for accident mitigation with safety margin to failure.

(2) Equipment that will experience environmental conditions of design-basis accidents through which it need not function for mitigation of such accidents but through which it must not fail in a manner detrimental to plant safety or accident mitigation must be qualified to demonstrate the capability to withstand any accident environment for the time during which it must not fail with safety margin to failure.

(3) Equipment that will experience environmental conditions of design-basis accidents through which it need not function for mitigation of such accidents and whose failure (in any mode) is deemed not detrimental to plant safety or accident mitigation need not be qualified for any accident

environment.

(4) Equipment that has performed its safety function prior to the exposure to an accident environment and whose failure (in any mode) is deemed not detrimental to plant safety and will not mislead the operator need not be qualified for an accident environment.

E-4 For each item of equipment in the categories of equipment listed in item E-3, provide the following:

(1) The system safety function requirements for equipment in categories E-3 (1), E-3 (2), and E-3 (3).

(2) An environmental envelope as a function of time that includes all extreme parameters, both maximum and minimum values, expected to occur during plant shutdown and design-basis accident (including LOCA and MSLB), including postaccident conditions, for equipment in categories E-3 (1) and E-3 (2).

(3) Length of time equipment in categories E-3 (1) and E-3 (2) must perform its safety function when subjected to any of the limiting environment specified above.

(4) The technical bases that justify the placement of each item of equipment in categories E-3 (2) E-3 (3) and E-3 (4).

E-5 For each item of equipment identified in categories E-3 (1) and E-3 (2), state the actual qualification envelope simulated during testing (defining the duration of the environment and the margin in excess of the design requirements). If any method other than type testing was used for qualification, identify the method and define the equivalent 'qualification envelope' so derived.

E-6 Provide a summary of test results that demonstrates the adequacy of the qualification program. If any analysis is used for qualification, justification of all analysis assumptions must be provided.

E-7 Identify the qualification documents that contain detailed supporting information, including test data, for items E-5 and E-6.