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DRAFT REGULATORY GUIDE

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(Proposed Revision 1 of Regulatory Guide 1.47, Rev. 0, dated May 1973)

BYPASSED AND INOPERABLE STATUS INDICATION FOR NUCLEAR POWER PLANT SAFETY SYSTEMS

A. INTRODUCTION

This guide describes a method that the staff of the U.S. Nuclear Regulatory Commission (NRC) considers acceptable for use in complying with the NRC's regulations with respect to bypassed and inoperable status indication for nuclear power plant safety systems.

The regulatory framework that the NRC has established for nuclear power plants consists of a number of regulations and supporting guidelines applicable to bypassed and inoperable status indication, including, but not limited to, General Design Criterion (GDC) 1, "Quality Standards and Records," GDC 13, "Instrumentation and Control," GDC 19, "Control Room," GDC 21, "Protection System Reliability and Testability," GDC 22, "Protection System Independence," and GDC 24, "Separation of Protection and Control Systems," as set forth in Appendix A, "General Design Criteria for Nuclear Power Plants," to Title 10, Part 50, "Domestic Licensing of Production and Utilization Facilities," of the *Code of Federal Regulations* (10 CFR Part 50) (Ref. 1). GDC 1 requires that structures, systems, and components important to safety be designed and installed to quality standards commensurate with the importance-to-safety of the functions to be performed. GDC 13 requires that appropriate controls be provided to maintain variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems within prescribed operating ranges. GDC 19 requires that a control room be provided from which actions can be taken to operate the nuclear power unit safely under normal operating conditions. GDC 21 requires that the protection system

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This regulatory guide is being issued in draft form to involve the public in the early stages of the development of a regulatory position in this area. It has not received final staff review or approval and does not represent an official NRC final staff position.

Public comments are being solicited on this draft guide (including any implementation schedule) and its associated regulatory analysis or value/impact statement. Comments should be accompanied by appropriate supporting data. Written comments may be submitted to the Rulemaking, Directives, and Editing Branch, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; e-mailed to nrcrep.resource@nrc.gov; submitted through the NRC's interactive rulemaking Web page at http://www.nrc.gov; faxed to (301) 415-5144; or hand-delivered to Rulemaking, Directives, and Editing Branch, Office of Administration, US NRC, 11555 Rockville Pike, Rockville, Maryland 20852, between 7:30 a.m. and 4:15 p.m. on Federal workdays. Copies of comments received may be examined at the NRC's Public Document Room, 11555 Rockville Pike, Rockville, MD. Comments will be most helpful if received by December 22, 2008.

Electronic copies of this draft regulatory guide are available through the NRC's interactive rulemaking Web page (see above); the NRC's public Web site under Draft Regulatory Guides in the Regulatory Guides document collection of the NRC's Electronic Reading Room at http://www.nrc.gov/reading-rm/doc-collections/; and the NRC's Agencywide Documents Access and Management System (ADAMS) at http://www.nrc.gov/reading-rm/adams.html, under Accession No. ML082140114.

be designed for high functional reliability and inservice testability. GDC 22 requires that the protection system be designed to ensure that the effects of normal operating, maintenance, and testing on redundant channels do not result in the loss of the protection function or be demonstrated to be acceptable on some other defined basis. GDC 24 requires that interconnection of the protection and control systems be limited to ensure that safety is not significantly impaired.

In 10 CFR 50.55a(h), the NRC requires compliance with Institute of Electrical and Electronics Engineers (IEEE) Standard (Std) 603-1991, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations," and the correction sheet dated January 30, 1995 (Ref. 2). For nuclear power plants with construction permits issued before January 1, 1971, the applicant/licensee may elect to comply instead with its plant-specific licensing basis. For nuclear power plants with construction permits issued between January 1, 1971, and May 13, 1999, the applicant/licensee may elect to comply instead with the requirements in IEEE Std 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations" (Ref. 3). Both IEEE Std 603-1991 and IEEE Std 279-1971 list requirements with regard to bypassed and inoperable status indication for safety systems. In addition, Criterion XIV, "Inspection, Test, and Operating Status," as given in Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50, requires that measures be established for indicating the operating status of structures, systems, and components of the nuclear power plant, such as by tagging valves and switches, to prevent inadvertent operation. The provisions of 10 CFR 50.34(f)(2)(v), "Additional TMI-Related Requirements," also require automatic indication of bypassed and operable status of safety systems.

IEEE Std 603-1991 uses the term "safety systems" rather than "protection systems" to define its scope. The standard offers the following definition of a "safety system":

> "…a system that is relied upon to remain functional during and following design basis events to ensure: (i) the integrity of the reactor coolant pressure boundary, (ii) the capability to shut down the reactor and maintain it in a safe shutdown condition, or (iii) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the 10 CFR Part 100 guidelines."

In addition, IEEE Std 603-1991 defines a "safety function" as "one of the processes or conditions (for example, emergency negative reactivity insertion, post-accident heat removal, emergency core cooling, post-accident radioactivity removal, and containment isolation) essential to maintain plant parameters within acceptable limits established for a design basis event." The standard also defines a "division" as "the designation applied to a given system or set of components that enable the establishment and maintenance of physical, electrical, and functional independence from other redundant sets of components."

IEEE Std 279-1971 states that a "protection system" encompasses all electric and mechanical devices and circuitry (from sensors to actuation device input terminals) involved in generating those signals associated with the protective function. These signals include those that actuate a reactor trip and that, in the event of a serious reactor accident, actuate engineered safety features, such as containment isolation, core spray, safety injection, pressure reduction, and air cleaning. This standard defines "protective function" as "the sensing of one or more variables associated with a particular generating station condition, signal processing, and the initiation and completion of the protective action at values of the variables established in the design bases."

Clause 4.13 of IEEE Std 279-1971 requires in part that, if the protective action of some part of the protection system has been bypassed or deliberately rendered inoperable for any purpose, this fact shall be

continuously indicated in the control room. Clause 5.8.3 of IEEE Std 603-1991 requires that, if the protective actions of some part of a safety system have been bypassed or deliberately rendered inoperative for any purpose other than an operating bypass, continued indication of this fact for each affected safety group shall be provided in the control room. Clause 5.8.3 also provides guidance that (1) the bypass display instrumentation need not be part of the safety systems, (2) this indication shall be automatically actuated if the bypass or inoperative condition is (a) expected to occur more frequently than once a year and (b) expected to occur when the affected system is required to be operable, and (3) the capability shall exist in the control room to manually activate this display indication.

The NRC issues regulatory guides to describe to the public 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 accidents, and to provide guidance to applicants. Regulatory guides are not substitutes for regulations and compliance with them is not required.
This regulatory guide contains information collection requirements covered by 10 CFR Part 50 that the Office of Management and Budget (OMB) approved under OMB control number 3150-0011. The NRC may neither conduct nor sponsor, and a person is not required to respond to, an information collection request or requirement unless the requesting document displays a currently valid OMB control number.

B. DISCUSSION

The Working Group Subcommittee 6.3 of the IEEE Nuclear Power Engineering Society developed IEEE Std 603-1991, and the IEEE Standards Board approved it on June 27, 1991. The standard provides guidance on the minimum functional design criteria for the electrical power, instrumentation, and control portions of nuclear power plant safety systems. This standard evolved from IEEE Std 279-1971 and, through interfaces with other referenced standards, reflects advances in digital technology. Nuclear power plants are currently replacing existing instrumentation and control (I&C) equipment with digital computer-based I&C systems or advanced analog systems. However, compared to existing I&C systems, these technologies may pose a number of new vulnerabilities for the nuclear power plant. This regulatory guide provides an acceptable method for establishing the design criteria for existing I&C systems and for establishing the design criteria for digital and advanced analog systems for bypassed and inoperable status indication.

Clause 5.8 of IEEE Std 603-1991 presents the requirements for safety system information displays. Clause 5.8.2 of IEEE Std 603-1991 requires that "display instrumentation shall provide accurate, complete, and timely information pertinent to safety system status." Clause 5.8.3 of the standard provides the specific guidance on the indication of bypasses, which may be provided by a nonsafety system. The clause states, "If the protective actions of some part of a safety system have been bypassed or deliberately rendered inoperative for any purpose other than an operating bypass, continued indication of this fact for each affected safety group shall be provided in the control room." IEEE Std 603-1991 defines a safety group as "a given set of interconnected components, modules, and equipment that can accomplish a safety function." Clause 5.8.3.2 of IEEE Std 603-1991 specifically requires that the indication of bypasses "shall be automatically actuated if the bypass or inoperative condition (a) is expected to occur more frequently than once a year and (b) is expected to occur when the affected system is required to be operable." Clause 5.8.4 of the standard requires that the location of information displays be "accessible to the operator."

Current designs of safety systems and protective actions are such that certain safety functions of a nuclear power plant may be bypassed or made inoperable during the performance of periodic tests or maintenance. Regulatory Guide 1.118, "Periodic Testing of Electric Power and Protection Systems" (Ref. 4), endorses IEEE Std 338-1987, "Criteria for the Periodic Testing of Nuclear Power Generating

Station Safety Systems" (Ref. 5), as an acceptable method for the periodic testing of electric power and protection systems. Generally, the plant's administrative procedures require that the operator give permission before the initiation of any activity that would or could affect a safety system. The decision to grant such permission should be based on knowledge of the operating status of the safety systems, the extent to which the activity will affect those systems, and whether that effect is permissible within the provisions of the license. Experience at operating plants, however, suggests that when the measures used to indicate inoperable status consist solely of administrative procedures, the operator is not always fully aware of the ramifications of each bypassed or inoperable component. Automatic indication display of any bypass or inoperability in a safety system supplements administrative procedures and aids the operator. Clause 5.16 of IEEE Std 338-1987 states that "indication should be provided in the control room if a portion of the safety system is inoperable or bypassed. Systems that are frequently placed in bypass or inoperative condition for the purposes of testing should have automatic indication."

Digital computer-based I&C systems make extensive use of self-testing. Digital computer-based I&C systems are more prone to different kinds of failures than traditional analog systems. Self-testing and watchdog timers should reduce the time to detect and identify failures. Computer self-testing is most effective at detecting random hardware failures. A bypass and inoperable status indication system should include a capability of assuring its operable status during normal plant operation to the extent that the indicating and annunciating function can be verified. However, the bypass and inoperable status indication system should be designed to avoid erroneous bypass indications. If the use of self-testing in digital computer-based I&C systems causes a protective action of some part of a safety system to be bypassed or deliberately rendered inoperative, then this fact should be automatically indicated in the control room.

With regards to surveillance testing, if the protective action of some part of a protection or safety system is bypassed or deliberately rendered inoperative for testing, that fact should be continuously indicated in the control room. Operations staff should also be able to confirm that a bypassed safety system has been properly returned to service. In a given plant design, it may be best to group the bypass indicators according to the safety systems' dependence on a common electric power supply; for example, the bypass indicators for all engineered safety feature systems that are assigned to one standby power source could be located near the bypass indicator for that source. Other groupings could be acceptable. The arrangement of bypass indicators should enable the operator to determine the status of each safety system and whether continued reactor operation is permissible. When a protective function of a shared system can be bypassed, the control room of each affected unit should receive an indication of that bypass condition. In any design, it may be necessary to include an audible, as well as visual, alarm to attract the operator's attention when the status of the safety system changes.

Clause 5.8.3.3 of IEEE Std 603-1991 requires that, for indication of bypasses, "the capability shall exist in the control room to manually activate this display indication." The effectiveness of an automatic indicating system is enhanced by including a manual capability to activate the indicators. Manual capability is useful in displaying those inoperable or bypassed conditions, whether deliberately induced or not, that are not automatically indicated.

Bypass indication should aid the operator in recognizing the effects on plant safety of seemingly unrelated or insignificant events. Therefore, the indication of bypass conditions should be at the safety group level, whether or not it is also at the channel, component, or module level. Clause 5.8.3.1 of IEEE Std 603-1991 states that "this display instrumentation need not be part of the safety systems." The indication system should be designed and installed in a manner that precludes the possibility of adverse effects on plant safety systems. The bypass and inoperable status indications may be provided via operator workstations or hardwired indicators and analog signal processing circuits.

However, Clause 5.6.3.1(1) of IEEE Std 603-1991 specifies that "equipment that is used for both safety and nonsafety functions shall be classified as part of the safety systems." Therefore, if the bypassed and inoperable status indication is used for both safety and nonsafety functions, then the indication system will require classification as part of safety systems. Unless the indication system is designed in conformance with criteria established for safety systems, it should not be used to perform functions that are essential to safety, and administrative procedures should not require immediate operator action based solely on bypass indications. Therefore, if the design or administrative procedures establish the bypassed and inoperable status indication as an element of the safety system, then the following paragraphs addressing single failure, independence, and qualification would apply.

If bypass and inoperable status indication is deemed to be part of a safety system, then the single-failure criterion of IEEE Std 603-1991, Clause 5.1, would apply to the indication system. Regulatory Guide 1.53, "Application of the Single-Failure Criterion to Safety Systems" (Ref. 6), endorses IEEE Std 379-2000, "IEEE Standard Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems" (Ref. 7), as an acceptable method to meet the regulations concerning the application of the single-failure criterion to the electrical power, instrumentation, and control portions of nuclear power plant safety systems. Digital system common-cause failures are not treated as a single failure with respect to the single-failure criterion. IEEE Std 7-4.3.2-2003, "IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations" (Ref. 8), provides guidance on using diversity to address common-cause failures in computer-based safety systems. In addition, NUREG/CR-6303, "Method for Performing Diversity and Defense-in-Depth Analyses of Reactor Protection Systems" (Ref. 9), describes a method for analyzing computer-based nuclear reactor protection systems to identify design vulnerabilities to common-mode failure.

In addition to meeting the single-failure criterion, if bypass and inoperable status indication is deemed to be part of a safety system, then maintaining independence between redundant portions of the safety system is essential to the effective utilization of the single-failure criterion. Regulatory Guide 1.75, "Independence of Electrical Safety Systems" (Ref. 10), provides guidance through the application of IEEE Std 384-1992, "IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits" (Ref. 11), to meet the regulations with respect to the physical independence requirements of the circuits and electric equipment that comprise or are associated with safety systems. Regulatory Guide 1.152, "Criteria for Digital Computers in Safety Systems of Nuclear Power Plants" (Ref. 12), endorses IEEE Std 7-4.3.2-2003 as an acceptable method for addressing high functional reliability and design requirements for computers used in safety systems of nuclear power plants, including safety-related digital communications, independence, and integrity.

If bypass and inoperable status indication is deemed to be part of a safety system, the equipment qualification criterion of IEEE Std 603-1991, Clause 5.4, would apply to the indication system. Clause 5.4 requires that safety system equipment be environmentally qualified. Regulatory Guide 1.209, "Guidelines for Environmental Qualification of Safety-Related Computer-Based Instrumentation and Controls Systems in Nuclear Power Plants" (Ref. 13), endorses IEEE Std 323-2003, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations" (Ref. 14), as acceptable guidance for environmental qualification of safety-related computer-based I&C systems for service in mild environments. Regulatory Guide 1.89, "Environmental Qualification of Certain Electronic Equipment Important to Safety for Nuclear Power Plants" (Ref. 15), provides guidance for environmental qualification of equipment intended for use in harsh environments. Regulatory Guide 1.180, "Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems" (Ref. 16), presents guidance for complying with the NRC's regulations on design, installation, and testing practices for addressing the effects of electromagnetic and radiofrequency interference (EMI/RFI) and power surges on

safety-related I&C systems. Clause 5.4 of IEEE Std 7-4.3.2-2003 provides requirements for equipment qualification of digital computers used in safety systems.

REGULATORY POSITION

Conformance with the requirements of IEEE Std 603-1991 provides a method that the NRC staff considers acceptable for satisfying the agency's regulatory requirements with respect to bypassed and inoperable status indication, subject to the following:

1. Administrative procedures should be supplemented by an indication system that automatically indicates for each affected safety system or subsystem the bypass or deliberately induced inoperability of a safety function and the systems actuated or controlled by the safety function. Provisions should also be made to allow operations staff to confirm that a bypassed safety function has been properly returned to service.

2. The indicating system of Position 1 above should also be activated automatically by the bypassing or deliberately induced inoperability of any auxiliary or supporting system that effectively bypasses or renders inoperable a safety function and the systems actuated or controlled by the safety function.

3. Annunciating functions for system failure and automatic actions based on self-test or self-diagnostic capabilities of digital computer-based I&C safety systems should be consistent with Positions 1 and 2 above.

4. The bypass and inoperable status indication system should include a capability for assuring its operable status during normal plant operation to the extent that the indicating and annunciating functions can be verified.

5. Bypass and inoperable status indicators should be arranged such that the operator can determine whether continued reactor operation is permissible. The control room of all affected units should receive indication of the bypass of shared system safety functions.

6. Bypass and inoperable status indicators should be designed and installed in a manner that precludes the possibility of adverse effects on plant safety systems. The indicator system should not be used to perform functions that are essential to safety, unless it is designed in conformance with criteria established for safety systems.

IMPLEMENTATION

The purpose of this section is to provide information to applicants and licensees regarding the NRC's plans for using this draft regulatory guide. The NRC does not intend or approve any imposition or backfit in connection with its issuance.

The NRC has issued this draft guide to encourage public participation in its development. The NRC will consider all public comments received in development of the final guidance document. In some cases, applicants or licensees may propose an alternative or use a previously established acceptable alternative method for complying with specified portions of the NRC's regulations. Otherwise, the methods described in this guide will be used in evaluating compliance with the applicable regulations for license applications, license amendment applications, and amendment requests.

REGULATORY ANALYSIS

1. Statement of the Problem

The current version of Regulatory Guide 1.47 (Revision 0), dated May 1973, is based solely on IEEE Std 279-1971. Section 50.55a, "Codes and Standards," of 10 CFR Part 50 requires that applications filed on or after May 13, 1999, for preliminary and final design approvals (Appendix O, "Standardization of Design: Staff Review of Standard Designs," to 10 CFR Part 52 (Ref. 17), "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants"), design certification, and construction permits, operating licenses, and combined licenses must meet the requirements for safety systems in IEEE Std 603-1991 and the correction sheet dated January 30, 1995. In addition, nuclear power plants are currently replacing existing I&C equipment with digital computer-based I&C systems or advanced analog systems. Regulatory Guide 1.47 (Revision 0) does not address the application of digital I&C systems to bypassed and inoperable status indication in nuclear power plants.
Therefore, revision of this regulatory guidance is necessary to incorporate the updated IEEE Std 603-1991 on safety systems endorsed by the NRC and to discuss the implications of the use of digital systems on the bypassed and inoperable status indication in nuclear power plants.

2. Objective

The objective of this regulatory action is to include appropriate references to IEEE Std 603-1991 and a discussion of digital computer-based I&C systems or advanced analog systems.

3. Alternative Approaches

The NRC staff considered the following alternative approaches:

- Do not revise Regulatory Guide 1.47.
- Update Regulatory Guide 1.47.

3.1 Alternative 1: Do Not Revise Regulatory Guide 1.47

Under this alternative, the NRC would not revise this guidance, and the original version of this regulatory guide would be retained. This alternative is considered the baseline or "no action" alternative and, as such, involves no value/impact considerations. This alternative would leave the regulatory guide outdated with the current regulations in 10 CFR 50.55a(h) and 10 CFR 50.34(f)(2)(v), and would not address the existing new digital I&C technology.

3.2 Alternative 2: Update Regulatory Guide 1.47

Under this alternative, the NRC would update Regulatory Guide 1.47 to reflect the requirements of IEEE Std 603-1991. The updated version would include a discussion of digital computer-based I&C systems and advanced analog systems. The update would retain references to IEEE Std 279-1971, because this IEEE standard remains applicable to plants with construction permits issued after January 1, 1971, but before May 13, 1999.

One benefit of this action is that it would enhance reactor safety by addressing the revised IEEE Std 603-1991 on safety systems endorsed by the NRC. Important digital computer-based I&C system issues such as single-failure criteria and communications would receive consideration. Also, the updated

guide would address considerations from current regulations in 10 CFR 50.55a(h), and 10 CFR 50.34(f)(2)(v).

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The impact to the NRC would be the costs associated with preparing and issuing the regulatory guide revision. The impact to the public would be the voluntary costs associated with reviewing and providing comments to NRC during the public comment period. The value to NRC staff and its applicants would be the benefits associated with enhanced efficiency and effectiveness in using a common guidance document as the technical basis for license applications and other interactions between the NRC and its regulated entities.

4. Conclusion

Based on this regulatory analysis, the NRC staff recommends revision of Regulatory Guide 1.47. The staff concludes that the proposed action will enhance reactor safety by reflecting current regulations and referencing the updated IEEE Std 603-1991on safety systems endorsed by the NRC. The revised guide could also lead to cost savings for the industry, especially with regard to applications for standard plant design certifications and combined licenses.

REFERENCES

1. 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," U.S. Nuclear Regulatory Commission, Washington, DC.

2. IEEE Std 603-1991, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers, Piscataway, NJ, 1991, and the correction sheet dated January 30, 1995.

3. IEEE Std 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers, Piscataway, NJ, 1971.

4. Regulatory Guide 1.118, "Periodic Testing of Electric Power and Protection Systems," U.S. Nuclear Regulatory Commission, Washington, DC, April 1995.

5. IEEE Std 338-1987, "Criteria for the Periodic Testing of Nuclear Power Generating Station Safety Systems," Institute of Electrical and Electronics Engineers, Piscataway, NJ, 1987.

6. Regulatory Guide 1.53, "Application of the Single-Failure Criterion to Safety Systems," U.S. Nuclear Regulatory Commission, Washington, DC, November 2003.

7. IEEE Std 379-2000, "IEEE Standard Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems," Institute of Electrical and Electronics Engineers, Piscataway, NJ, 2000.

8. IEEE Std 7-4.3.2-2003, "IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers, Piscataway, NJ, 2003.

9. NUREG/CR-6303, "Method for Performing Diversity and Defense-in-Depth Analyses of Reactor Protection Systems," U.S. Nuclear Regulatory Commission, Washington, DC, December 1994.

10. Regulatory Guide 1.75, "Independence of Electrical Safety Systems," U.S. Nuclear Regulatory Commission, Washington, DC, February 2005.

11. IEEE Std 384-1992, "IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits," Institute of Electrical and Electronics Engineers, Piscataway, NJ, 1992.

12. Regulatory Guide 1.152, "Criteria for Digital Computers in Safety Systems of Nuclear Power Plants," U.S. Nuclear Regulatory Commission, Washington, DC, January 2006.

13. Regulatory Guide 1.209, "Guidelines for Environmental Qualification of Safety-Related Computer-Based Instrumentation and Controls Systems in Nuclear Power Plants," U.S. Nuclear Regulatory Commission, Washington, DC, March 2007.

14. IEEE Std 323-2003, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers, Piscataway, NJ, 2003.

15. Regulatory Guide 1.89, "Environmental Qualification of Certain Electronic Equipment Important to Safety for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, Washington, DC, June 1984.

16. Regulatory Guide 1.180, "Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems," U.S. Nuclear Regulatory Commission, Washington, DC, October 2003.

17. 10 CFR Part 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, Washington, DC.