[7590-01-P]

NUCLEAR REGULATORY COMMISSION

10 CFR Part 50

[NRC-2011-0089]

RIN 3150-AI98

Incorporation by Reference of Institute of Electrical and Electronics Engineers Standard 603-2009

AGENCY: Nuclear Regulatory Commission.

ACTION: Proposed rule; request for comments.

SUMMARY: The U.S. Nuclear Regulatory Commission (NRC) proposes to amend its regulations to incorporate by reference the Institute of Electrical and Electronics Engineers Standard (IEEE Std) 603-2009, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations." The IEEE Std 603-2009 is the most recent version of IEEE Std 603 which addresses the power, instrumentation, and control systems for nuclear power reactors. Under the proposed amendment, design approvals and design certifications issued after the effective date of any final rule would be required to comply with IEEE Std 603-2009. Applications for construction permits submitted after the effective date of any final rule would be required to comply with IEEE Std 603-2009. Applications for license amendments for combined licenses and currently operating nuclear power plants received after the effective date of any final rule

may be required, to the extent applicable as specified in the proposed rule, to comply with IEEE Std 603-2009. The NRC is also making available for comment, as part of this notice, the draft guidance for the implementation of this proposed rule.

DATES: Submit comments by **[INSERT DATE 120 DAYS AFTER PUBLICATION IN THE *FEDERAL REGISTER*]**. Comments received after this date will be considered if it is practical to do so, but the NRC is able to ensure consideration of comments received only on or before this date.

PUBLIC MEETINGS: The U.S. Nuclear Regulatory Commission intends to conduct one or more public workshops on the proposed rulemaking during the public comment period; refer to the NRC's public meeting schedule on the NRC Web site, http://www.nrc.gov or directly at http://meetings.nrc.gov/pmns/mtg.

ADDRESSES: You may submit comments by any of the following methods (unless this document describes a different method for submitting comments on a specific subject):

- **Federal rulemaking Web Site:** Go to http://www.regulations.gov and search for Docket ID NRC-2011-0089. Address questions about NRC dockets to Carol Gallagher; telephone: 301-415-3463; e-mail: Carol.Gallagher@nrc.gov. For technical questions contact the individuals listed in the FOR FURTHER INFORMATION CONTACT section of this proposed rule.
- **E-mail comments to:** Rulemaking.Comments@nrc.gov. If you do not receive an automatic e-mail reply confirming receipt, then contact us at 301-415-1677.
- **Fax comments to:** Secretary, U.S. Nuclear Regulatory Commission at 301-415-1101.
- **Mail comments to:** Secretary, U.S. Nuclear Regulatory Commission, Washington, DC

20555-0001, ATTN: Rulemakings and Adjudications Staff.

- **Hand deliver comments to:** 11555 Rockville Pike, Rockville, Maryland 20852, between 7:30 a.m. and 4:15 p.m. (Eastern Time) Federal workdays; telephone: 301-415-1677.

For additional direction on obtaining information and submitting comments, see "Obtaining Information and Submitting Comments" in the SUPPLEMENTARY INFORMATION section of this document.

FOR FURTHER INFORMATION CONTACT: Daniel I. Doyle, Office of Nuclear Reactor Regulation, telephone: 301-415-3748, e-mail: Daniel.Doyle@nrc.gov or Richard J. Stattel, Office of Nuclear Reactor Regulation, telephone: 301-415-8472, e-mail: Richard.Stattel@nrc.gov. Both are staff of the U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001.

SUPPLEMENTARY INFORMATION:

EXECUTIVE SUMMARY:

A. Need for the Regulatory Action

The NRC is proposing to amend its regulations to incorporate by reference a voluntary consensus standard, IEEE Std 603-2009, to establish functional and design requirements for power, instrumentation, and control systems for nuclear power plants. This action would be consistent with the provisions of the National Technology Transfer and Advancement Act of 1995, Pub. L. 104-113 (NTTAA), which encourage Federal regulatory agencies to consider adopting voluntary consensus standards as an alternative to agency development of government-unique standards. This action also would be consistent with NRC's practice to evaluate the latest version of a voluntary consensus standard for its suitability for endorsement by regulation or regulatory guidance. The final rule would become effective 30 days after publication in the *Federal Register* (FR).

B. Major Provisions

The proposed rule would incorporate by reference the latest version of IEEE Std 603 which addresses the functionality and design requirements for power, instrumentation, and control systems for nuclear power plants.

- The proposed rule defines the conditions that would allow existing licensees to replace plant equipment while maintaining its existing licensing basis with respect to power, instrumentation, and control systems.
- The proposed rule defines the conditions for which existing permit, license, certificate, standard design, and standard design approvals would be required to address the new standard in modifications and applications related to power, instrumentation, and control systems.
- The proposed rule imposes conditions upon the use of IEEE Std 603-2009 in the areas of system integrity, independence, maintenance bypass, and maintenance of records for power, instrumentation, and control systems.

C. Costs and Benefits

The NRC prepared a draft regulatory analysis to examine the costs and benefits of the alternatives considered by the NRC. Among the other quantitative factors, the draft regulatory analysis qualitatively considered factors including regulatory efficiency and consistency with the NTTAA which directs Federal agencies to adopt voluntary consensus standards whenever possible.

The analysis concluded that the proposed rule relative to the regulatory baseline is cost-benefit neutral for industry with an estimate net cost of $7,000 based on a 7-percent net present value to a net benefit of $26,000 based on a 3-percent net present value. For the NRC, the proposed rule is not quantitatively cost beneficial, although, as discussed below, there are significant benefits that were not quantified in this analysis. The quantified costs for the NRC range from an estimated net cost of $372,000 based on a 7% net present value to a net cost of $355,000 based on a 3% net present value. The NRC benefits from the proposed rulemaking because of the averted cost savings resulting from the reduction in the number of alternative requests on a plant-specific basis under 10 CFR 50.55a(z).

The proposed rule has the qualitative benefit of meeting the NRC goal of ensuring the protection of public health and safety and the environment through the NRC's approval of the criteria in IEEE Std 603-2009 to address safety issues associated with major changes to the underlying bases of protection and safety systems that could impair dependability and reliability from potential new system-level failure modes. Based on experience, the NRC staff believes that the improvements provided by the proposed rule would reduce this level of industry operation impact and provide additional averted costs for the preparation of additional supplements and for responding to Request for Additional Information, both of which were not quantified.

The proposed rule creates a regulatory framework that could accelerate the pace at which licensees upgrade nuclear plant instrumentation and control (I&C) systems. The rule would provide regulatory certainty for upgrading systems from analog instrumentation to digital instrumentation allowing licensees to take advantage of the benefits of these digital system upgrades. These benefits include operation and maintenance cost reduction through decreased obsolescence, fewer licensee event reports, additional performance benefits, and increased safety.

If the quantified costs and benefits were considered in isolation, the NRC would not proceed with this rulemaking because the total quantified benefits of the proposed regulatory action do not equal or exceed the costs of the proposed action. However, it is the NRC's proposed determination that the values (including the safety benefit, averted cost savings, and other non-quantified benefits), considered together, outweigh the identified impacts. For more information, please see the draft regulatory analysis (Agencywide Documents Access Management System (ADAMS) Accession No. ML120310194). The NRC is requesting public comment on the draft regulatory analysis.

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I. Obtaining Information and Submitting Comments.

A. Obtaining Information.

Please refer to Docket ID NRC-2011-0089 when contacting the NRC about the availability of information for this action. You may obtain publicly-available information related to this action by any of the following methods:

- **Federal rulemaking Web Site:** Go to http://www.regulations.gov and search for Docket ID NRC-2011-0089.

- **NRC's Agencywide Documents Access and Management System (ADAMS):** You may obtain publicly-available documents online in the ADAMS Public Documents collection at http://www.nrc.gov/reading-rm/adams.html. To begin the search, select "Begin Web-Based ADAMS Search." For problems with ADAMS, please contact the NRC's Public Document Room (PDR) reference staff at 1-800-397-4209, 301-415-4737, or by e-mail to pdr.resource@nrc.gov. For the convenience of the reader, instructions about accessing documents referenced in this document are provided in the "Availability of Documents" section.

- **NRC's PDR:** You may examine and purchase copies of public documents at the NRC's PDR, Room O1-F21, One White Flint North, 11555 Rockville Pike, Rockville, Maryland 20852.

B. Submitting Comments.

Please include Docket ID NRC-2011-0089 in the subject line of your comment submission, in order to ensure that the NRC is able to make your comment submission available to the public in this docket.

The NRC cautions you not to include identifying or contact information in comment submissions that you do not want to be publicly disclosed in your comment submission. The NRC will post all comment submissions at http://www.regulations.gov as well as enter the comment submissions into ADAMS, and the NRC does not routinely edit comment submissions to remove identifying or contact information.

If you are requesting or aggregating comments from other persons for submission to the NRC, then you should inform those persons not to include identifying or contact information that they do not want to be publicly disclosed in their comment submission. Your request should state that the NRC does not routinely edit comment submissions to remove such information before making the comment submissions available to the public or entering the comment into ADAMS.

II. Background.

It has been the NRC's practice to establish requirements for the protection systems and safety systems in nuclear power plants by incorporating by reference certain standards published by the IEEE into § 50.55a of Title 10 of the *Code of Federal Regulations* (10 CFR), "Domestic Licensing of Production and Utilization Facilities."

Paragraph 50.55a(h)(2), "Protection systems," currently requires that the protection systems in nuclear power plants with construction permits issued after January 1, 1971, but before May 13, 1999, meet the requirements stated in either IEEE Std 279, "Criteria for

Protection Systems for Nuclear Power Generating Stations," or in IEEE Std 603-1991, "IEEE Criteria for Safety Systems for Nuclear Power Generating Stations," and the correction sheet dated January 30, 1995. For nuclear power plants with construction permits issued before January 1, 1971, § 50.55a(h)(2) requires that protection systems must be consistent with their licensing basis or meet the requirements of IEEE Std 603-1991 and the correction sheet dated January 30, 1995.

Paragraph 50.55a(h)(3), "Safety systems," currently requires that applications filed on or after May 13, 1999, for construction permits and operating licenses under 10 CFR part 50, standard design approvals, standard design certifications, and combined licenses under 10 CFR part 52 meet the requirements for safety systems stated in IEEE Std 603–1991 and the correction sheet dated January 30, 1995.

The IEEE has superseded the previous standards with IEEE Std 603-2009, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations," and the correction sheet dated March 10, 2015. The proposed rule would update the current NRC regulations to incorporate by reference this standard and to specify requirements for using this latest version of IEEE Std 603 or earlier versions of this standard on the basis of license date, construction permit date, and type of protection system or safety system modification. This proposed rule would apply to: 1) reactor design applications for a license, construction permit, design approval, or design certification, and 2) applications for license amendments for nuclear power plants. A final rule would become effective 30 days after publication in the FR.

On August 4, 2015, the NRC staff held a public meeting to discuss the preliminary draft proposed rule language with interested members of the public. The NRC staff discussed the provisions of the draft proposed rule and solicited informal feedback. The staff's presentations slides and a summary of the meeting are available under ADAMS Accession No. ML15216A636.

III. Discussion.

A. IEEE Std 603-2009.

In publishing IEEE Std 603-2009, the IEEE departed from the approach in IEEE Std 603-1991. The IEEE Std 603-2009: 1) addresses potential safety issues that might arise from incorporating components that use advanced technologies in safety systems; 2) contains additional and updated references and does not include references that are no longer in effect; 3) provides guidance to address electromagnetic compatibility issues; 4) adds new guidance on common cause failure; 5) contains classification requirements for equipment not credited to perform a safety function but is connected to safety-related equipment; 6) removes the requirement in section 6.7, “Maintenance bypass,” for meeting the single failure criterion during maintenance activities; and 7) specifically requires electrical isolation and digital communication independence between safety systems and non-safety systems. Consequently, the NRC proposes to update § 50.55a to incorporate by reference IEEE Std 603-2009 and the correction sheet dated March 10, 2015, with conditions, in addition to retaining the incorporation by reference for IEEE Std 279-1968, IEEE Std 279-1971, IEEE Std 603-1991, and the IEEE Std 603-1991 correction sheet dated January 30, 1995.

This proposed rule would incorporate a voluntary consensus standard, IEEE Std 603-2009 and the correction sheet dated March 10, 2015, into the NRC regulations to establish functional and design requirements for power, instrumentation, and control systems for nuclear power plants. This action would be consistent with the provisions of the NTTAA, which encourage Federal regulatory agencies to consider adopting voluntary consensus standards as an alternative to agency development of government-unique standards. This action also would be consistent with the NRC’s practice to evaluate the latest version of a consensus standard for its suitability for endorsement by regulation or regulatory guidance.

The development of IEEE voluntary consensus standards and the incorporation of the resulting standards into the NRC regulations is a three-step process. First, the IEEE establishes a group of stakeholders with varied viewpoints and interests to develop guidance and criteria in a specific topic area. This group often includes NRC staff representing the NRC's interests. This group then develops a draft standard or revises an existing standard to address a specific area of interest. In this drafting process, the group develops criteria, guidance, and technical justifications to address the draft standard's scope. Upon completion of the drafting process, the group submits the draft standard to the IEEE for the next step of the standard development process.

Second, the IEEE creates a ballot pool of stakeholders, which often includes NRC staff, to review the draft standard, vote to approve the draft standard, or provide to the IEEE comments and suggested revisions that could result in an approvable standard. Of this ballot pool, 75 percent or more must respond either in the affirmative (approve), in the negative (do not approve), or in abstention (choose not to vote), and there must be less than 30 percent abstentions from this ballot pool. Of the 75 percent or more respondents, 75 percent or more of the respondents must approve the standard (i.e., respond in the affirmative). Objections and comments submitted by the ballot pool respondents are considered by the IEEE (typically by the working group members) and are resolved to the extent that a consensus for publishing the standard can be obtained. All members of the balloting group are given an opportunity to see all the unresolved negative comments and the reasons why these comments could not be resolved. The balloting group members also are given the opportunity to change their votes as a result of change(s) made to resolve the negative ballots. The IEEE then publishes the approved standard.

Third, the NRC reviews the published IEEE standard to determine its acceptability for incorporation by reference in the NRC regulations or for use as guidance in regulatory activities.

The rulemaking process, when considered together with the IEEE process for developing and approving IEEE standards, constitutes the NRC's basis for determining that an IEEE standard (with conditions, as necessary) provides criteria upon which the NRC can conclude there is reasonable assurance of adequate protection of public health and safety and the environment.

The NRC reviewed IEEE Std 603-2009 and the correction sheet dated March 10, 2015, and concludes, under the process for reviewing IEEE standards, that, with conditions on its application, this standard is technically adequate, is consistent with current NRC regulatory policy, and should be used to specify regulatory criteria.

Currently, § 50.55a(h)(2), "Protection systems," specifies that the protection systems in nuclear power plants with construction permits issued after January 1, 1971, but before May 13, 1999, must meet the requirements stated either in 1) IEEE Std 279, "Criteria for Protection Systems for Nuclear Power Generating Stations," or in 2) IEEE Std 603-1991, "[IEEE Standard] Criteria for Safety Systems for Nuclear Power Generating Stations," and the IEEE Std 603-1991 correction sheet dated January 30, 1995, "IEEE [Standard] Criteria for Safety Systems for Nuclear Power Generating Stations Correction Sheet." For nuclear power plants with construction permits issued before January 1, 1971, § 50.55a(h)(2) requires that protection systems must be consistent with their licensing basis or may meet the requirements of IEEE Std 603-1991 and the correction sheet dated January 30, 1995.

Further, § 50.55a(h)(3), "Safety systems," currently specifies that applications filed on or after May 13, 1999, for construction permits and operating licenses under 10 CFR part 50 and for standard design certifications, and combined licenses under 10 CFR part 52 must meet the requirements for safety systems in IEEE Std 603-1991 and the IEEE Std 603-1991 correction sheet dated January 30, 1995.

The 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 reactor trips and that, following certain events, actuate engineered safeguards, such as containment isolation, core spray, safety injection, pressure reduction, and air cleaning. "Protective function" is defined in IEEE Std 279-1971 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."

The IEEE Std 603-1991 and IEEE Std 603-2009 use the term "safety system" rather than "protection system." A "safety system" is defined in IEEE Std 603-1991 (and in IEEE Std 603-2009) as:

> [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 off-site exposures comparable to the 10 CFR Part 100 guidelines.

A "safety system" is considered a minimum set of interconnected components, modules, signal processors, and equipment that is relied upon to accomplish one or more safety functions (e.g., equipment relied upon to remain functional during and following design basis accidents). Safety system is a broad-based and all-encompassing term, embracing the protection system in addition to other electrical systems. Thus, the term "protection system" is not synonymous with the term "safety system," but instead is a subset of "safety systems." Nuclear power plant protection systems and safety systems are identified in the plant's final safety analysis report (FSAR).

In the context of § 50.55a(h), the term "current reactors" means nuclear power plants whose construction permits were issued before May 13, 1999.

In the context of § 50.55a(h), the term "data communication" means a method of transmitting and receiving information in which the information is encoded in a specific format (e.g., header, data content, and end of message) using software.

In the context of § 50.55a(h), the term "defense-in-depth" means an approach to designing and operating nuclear facilities that prevents and mitigates accidents that release radiation or hazardous materials. The key is multiple independent and redundant layers of defense to compensate for potential human and mechanical failures so that no single layer, no matter how robust, is relied upon exclusively. The defense-in-depth design approach includes the use of access controls, physical barriers, redundant and diverse key safety functions, and emergency response measures. More succinctly, "defense-in-depth," in the context of § 50.55a(h), means the principle of using different functional barriers to the propagation of faults to compensate for failures in other barriers.

In the context of § 50.55a(h), the term "diversity" means the use of different means including function, design, principles of operation, and organizational and development strategies to compensate for failures within a safety system.

Protection system and safety system diversity strategies use different means to compensate for failures within the protection system and safety system. Defense-in-depth strategies use different functional barriers (e.g., a non-safety control system and a reactor trip system) to compensate for potential failures in other functional barriers. Implementation of defense-in-depth and diversity strategies assure protection system and safety system independence from coincident failures or propagated failures due to the effects of natural phenomena, normal operation, postulated functional barrier failure modes, maintenance, testing, and postulated accident conditions.

In the context of § 50.55a(h), the term "function" means a specific process, action, or task that a system is to perform. More specifically, the term "function" is the process by which inputs into a structure, system, or component are transferred to outputs from the structure, system or component by some mechanism and, subject to certain controls, that can be identified by a function name and can be modeled as a unique entity. For example, a reactor trip system function consists of the reactor process measurement instrumentation, the reactor trip logic processing components, the reactor trip breakers, and the medium by which the input signals, the logic processing signals, and the output signals are transmitted to components in the safety function process (i.e., inputs, processing, outputs, and actuated devices).

In the context of § 50.55a(h), the term “functionality” means the set of functions or capabilities associated with software, computer hardware, or a component. These functions include the safety functions needed to actuate safety equipment and supporting features that are not required to perform the safety function, such as self-testing and diagnostic features and human-system interface functions.

In the context of § 50.55a(h), the term “hardwired connections” means a permanent physical point-to-point connection that is used to transmit signals. Hardwired connections can be implemented using various physical media (e.g., copper wire and optical fiber).

In the context of § 50.55a(h), the term “new reactors” means design certifications; standard design approvals; manufacturing licenses; and combined licenses not referencing a design certification, standard design approval, or manufacturing license under 10 CFR part 52 issued on or after the effective date of the final rule; construction permits and operating licenses under 10 CFR part 50 issued on or after the effective date of the final rule, except for an applicant for an operating license who received a construction permit for that facility before the effective date of the final rule; and holders of combined licenses issued under 10 CFR part 52 before the effective date of the final rule, but only if the combined license holder voluntarily modifies its data communication independence strategy.

In the context of § 50.55a(h), the term “physical mechanism” means a means to enforce one way communication from safety systems to non-safety systems through a hardware-based method such that no software is used to maintain the direction of data flow.

In the context of § 50.55a(h), the term “predictable” means the ability to determine the output of a system at any time through known relationships among the controlled system states and required responses to those states, such that a given set of input signals will always produce the same output signals.

In the context of § 50.55a(h), the term “repeatable” means the output of a system being consistently achieved given the same input and system properties (including internal and external conditions).

In the context of § 50.55a(h), the term “safety benefit” means a justification for adding safety system functionality that is not necessary to accomplish a safety function, but that contributes to safety (e.g., by increasing safety system availability or increasing the safety of a mechanical, nuclear, or electrical system design).

In the context of § 50.55a(h), the term “safety function” means 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 functional portion of a safety system consists of those functions of a safety system that must operate correctly for the safety system to accomplish its safety function.

In the context of § 50.55a(h), the term “safety system function” means any function performed by the safety system, including safety functions and other functions.

In the context of § 50.55a(h), the term “signal” means a detectable and measurable representation of a physical quantity by which messages or information can be transmitted. Signals can either be digital or analog in nature.

In the context of § 50.55a(h), the term “signal sharing” means the replication or duplication of a signal in one system and subsequent transmission to a different system. Signals can be shared through various media, including copper wires and optical links.

In the context of § 50.55a(h)(5)(iii)(C), the term “support(s) safety” means activities or functions that are necessary to accomplish a safety function or prevent impairment of a safety function.

In the context of § 50.55a(h), the term “technology” means the methods, techniques, and materials that are used to develop and implement a protection system function or a safety system function. For example, differences in technology exist in the methods, techniques, and materials for implementing a safety function with analog technology, microprocessor technology, and field programmable gate array (FPGA) technology. These technologies are significantly different from one another in system development processes, format of the function logic (e.g., arrangement of discrete electronic components versus software versus hardware description language, respectively), supporting hardware components, and operating and maintenance characteristics. The safety issues arising from these differences in characteristics between technologies could be sufficiently different that a licensee or applicant could be challenged to address issues such as electromagnetic compatibility (EMC), equipment qualification (EQ), common cause failure mitigation, and digital communication independence. Converting an analog-based safety function or system into a microprocessor-based safety function or system, and replacing a microprocessor-based safety function or system with an FPGA-based safety function or system are two examples of technology changes.

Paragraph 50.55a(h)(1) would be revised to include definitions for the terms “current reactors” and “new reactors” in the context of § 50.55a(h).

Conditions for the use of IEEE Std 279 and versions of IEEE Std 603 would be provided in § 50.55a(h)(2)(i) through (vii) to clarify for protection systems and safety systems the applicability of IEEE Std 603-2009 and earlier standards requirements for operating plants, new plants, and manufacturing licenses on the basis of the issue date of the construction permit, standard design certification, or manufacturing license. The regulatory requirements in § 50.55a(h)(2) would also reduce uncertainty and improve efficiency by identifying the specific criteria to be addressed for protection systems and safety systems. The following discussion addresses the basis underlying each of the conditions under § 50.55a(h)(2).

Proposed § 50.55a(h)(2)(i) would be added to clarify the requirements for protection systems and safety systems in nuclear power plants with construction permits issued before January 1, 1971. Licensees of plants in this category would be allowed to retain the licensing basis of their plant protection systems and safety systems (i.e., the plant licensing basis or IEEE Std 603-1991 and the correction sheet dated January 30, 1995). Licensees would not be required to modify or replace protection systems or safety systems to meet the requirements in IEEE Std 603-2009 and the correction sheet dated March 10, 2015. This paragraph is not intended to allow licensees to lessen the requirements stated in their existing protection system or safety system licensing basis. For example, a safety system that meets the requirements stated in IEEE Std 603-1991 and the correction sheet dated January 30, 1995, could not be modified such that it met only the requirements stated in its original licensing basis. By preserving the current licensing basis for the protection systems and safety systems addressed in this paragraph, licensees would not be required to modify or replace systems that were approved prior to the effective date of this rule to meet the requirements stated in IEEE Std 603-2009 and the correction sheet dated March 10, 2015. However, licensees would have the option to change the licensing basis of their plant protection systems and safety systems to meet the requirements stated in IEEE Std 603-2009 and the correction sheet dated March 10, 2015, subject to the conditions in § 50.55a(h)(3) through (8).

Proposed § 50.55a(h)(2)(ii) would be added to clarify the requirements for protection systems and safety systems in nuclear power plants whose construction permits were issued on or after January 1, 1971, but before May 13, 1999. This paragraph does not apply to combined licenses for standard design certifications. Protection systems and safety systems that are not subject to the requirements of § 50.55a(h)(3) would be required to meet the requirements stated in the protection system or safety system licensing basis after the effective date of this rule instead of the requirements stated in IEEE Std 603-2009 and the correction sheet dated March 10, 2015 (i.e., IEEE Std 279-1968, IEEE Std 279-1971, or IEEE Std 603-1991 and the IEEE Std 603-1991 correction sheet dated January 30, 1995). This paragraph is not intended to allow licensees to lessen the requirements stated in the licensing basis for their protection systems or safety systems. For example, a safety system whose current licensing basis is IEEE Std 603-1991 and the IEEE Std 603-1991 correction sheet dated January 30, 1995, could not be modified such that it met only the protection system requirements stated in IEEE Std 279-1971. By preserving the current licensing basis for the plant protection systems and safety systems addressed in this paragraph, licensees would not be required to modify or replace systems that were approved prior to the effective date of this rule to meet the safety system requirements stated in IEEE Std 603-2009 and the correction sheet dated March 10, 2015. However, licensees would have the option to meet the safety system requirements stated in IEEE Std 603-2009 and the correction sheet dated March 10, 2015, subject to the conditions in § 50.55a(h)(3) through (8), when modifying or installing protection systems and safety systems. No construction permits were issued between May 13, 1999, and the effective date of this rule.

Proposed § 50.55a(h)(2)(iii) would be added to clarify the requirements for protection systems and safety systems in standard design certifications issued after January 1, 1971, but before May 13, 1999. Two standard design certifications have been codified in 10 CFR part 52 between these dates: the U.S. Advanced Boiling Water Reactor (ABWR) (10 CFR part 52, appendix A) and the System 80+ (10 CFR part 52, appendix B). As specified in §§ 52.63, 52.83, 52.98, and 52.171, subject to the requirements stated in § 50.55a(h)(3), the protection systems in these two standard design certifications are required to meet the requirements stated in IEEE Std 279-1971 instead of the requirements stated in IEEE Std 603-2009 and the correction sheet dated March 10, 2015, regardless of the date a combined license referencing either standard design certification plant is issued. For example, an applicant obtaining a combined license for an ABWR nuclear power plant would be required to meet the protection system requirements stated in IEEE Std 279-1971 instead of the safety system requirements stated in IEEE Std 603-2009 and the correction sheet dated March 10, 2015, even if the combined license is issued after the effective date of this rule.

Proposed § 50.55a(h)(2)(iv) would be added to clarify the requirements for safety systems in standard design certifications issued on or after May 13, 1999, but before the effective date of this rule. As of April 1, 2015, three standard design certifications have been codified in 10 CFR part 52 after May 13, 1999: 1) a 600 MWe advanced pressurized water reactor (the AP600) (10 CFR part 52, appendix C), 2) a 1,000 MWe advanced pressurized water reactor (the AP1000) (10 CFR part 52, appendix D), and 3) a 1,600 MWe advanced boiling water reactor (the ESBWR) (10 CFR part 52, appendix E). As specified in §§ 52.63, 52.83, 52.98, and 52.171, subject to the requirements in § 50.55a(h)(3), the safety system designs in these three standard design certifications are required to meet the requirements stated in IEEE Std 603-1991 and the IEEE Std 603-1991 correction sheet dated January 30, 1995, instead of the requirements stated in IEEE Std 603-2009 and the correction sheet dated March 10, 2015. For example, an applicant applying after the effective date of this rule for a combined license for an AP1000 nuclear power plant would be required to meet the requirements stated in IEEE Std 603-1991 and the correction sheet dated January 30, 1995, instead of the requirements stated in IEEE Std 603-2009 and the correction sheet dated March 10, 2015, even if the combined license would be issued after the effective date of this rule.

Proposed § 50.55a(h)(2)(v) would be added to clarify the safety system requirements for standard design certifications issued after the effective date of this rule. Safety systems in standard design certifications issued after the effective date of this rule would be required to meet the requirements stated in IEEE Std 603-2009 and the correction sheet dated March 10, 2015, subject to the conditions in § 50.55a(h)(4) through (8).

Proposed § 50.55a(h)(2)(vi) would be added to clarify the requirements for protection system designs and safety system designs for nuclear power plants with construction permit applications under 10 CFR part 50 submitted after the effective date of this rule. The protection system designs and safety system designs in construction permit applications under 10 CFR part 50 submitted after the effective date of this rule would be required to meet the requirements stated in IEEE Std 603-2009 and the correction sheet dated March 10, 2015, subject to the conditions in § 50.55a(h)(3) through (8).

Proposed § 50.55a(h)(2)(vii) would be added to clarify the requirements for safety system designs in nuclear power plant combined licenses and manufacturing licenses under 10 CFR part 52 issued after the effective date of this rule. Combined licenses and manufacturing licenses that reference a standard design certification issued before the effective date of this rule would be required to meet the requirements stated in the referenced standard design certification. For example, a safety system design for a combined license issued after the effective date of this rule that referenced a standard design certification issued on or after May 13, 1999, but before the effective date of this rule would be required to meet the requirements stated in IEEE Std 603-1991 and the IEEE Std 603-1991 correction sheet dated January 30, 1995, instead of meeting the requirements stated in IEEE Std 603-2009 and the correction sheet dated March 10, 2015. Safety system designs in combined licenses and manufacturing licenses that reference a standard design certification issued after the effective date of this rule would be required to meet the requirements stated in IEEE Std 603-2009 and the correction sheet dated March 10, 2015, subject to the conditions in § 50.55a(h)(3) through (8).

Table 1 summarizes the proposed § 50.55a(h)(2) criteria to be met on the basis of the issue date of a plant's construction permit under 10 CFR part 50 and standard design certification, combined license, or manufacturing license under 10 CFR part 52. The standards listed in the "Standard Applicability" column designate the licensing basis standards that would be applicable for the corresponding § 50.55a paragraph. References to IEEE Std 603-1991 include the IEEE Std 603-1991 correction sheet dated January 30, 1995. References to IEEE Std 603-2009 include the IEEE Std 603-2009 correction sheet dated March 10, 2015.

Table 1 - 10 CFR 50.55a(h)(2) Issue date applicability.

<table>
<tr><th>Construction Permit, Standard Design Certification, Combined License, or Manufacturing License Issue Date</th><th>10 CFR 50.55a Paragraph</th><th>Standard Applicability</th></tr>
<tr><td>Nuclear power plant construction permits issued before January 1, 1971.</td><td>(h)(2)(i)</td><td>Licensing Basis
IEEE Std 603-1991</td></tr>
<tr><td>Nuclear power plant construction permits issued on or after January 1, 1971, but before May 13, 1999.</td><td>(h)(2)(ii)</td><td>IEEE Std 279-1968
IEEE Std 279-1971
IEEE Std 603-1991</td></tr>
<tr><td>Standard design certifications issued before May 13, 1999.</td><td>(h)(2)(iii)</td><td>IEEE Std 279-1971</td></tr>
<tr><td>Standard design certifications issued on or after May 13, 1999, but before [EFFECTIVE DATE OF THE FINAL RULE].</td><td>(h)(2)(iv)</td><td>IEEE Std 603-1991</td></tr>
<tr><td>Standard design certifications issued after [EFFECTIVE DATE OF THE FINAL RULE].</td><td>(h)(2)(v)</td><td rowspan="2">IEEE Std 603-2009</td></tr>
<tr><td>Applications submitted after [EFFECTIVE DATE OF THE FINAL RULE] for nuclear power plant construction permits under 10 CFR part 50.</td><td>(h)(2)(vi)</td></tr>
</table>

Nuclear power plant combined licenses and manufacturing licenses under 10 CFR part 52 issued after **[EFFECTIVE DATE OF THE FINAL RULE].**	**(h)(2)(vii)** Referenced SDC[1] issued before **[EFFECTIVE DATE OF THE FINAL RULE].**	IEEE Std 279-1971 IEEE Std 603-1991
	(h)(2)(vii) Referenced SDC issued after **[EFFECTIVE DATE OF THE FINAL RULE].**	IEEE Std 603-2009

Conditions for meeting the criteria stated in IEEE Std 279 and versions of IEEE Std 603 have been proposed in § 50.55a(h)(3) to clarify the applicability of IEEE Std 603-2009 and earlier standards for currently operating plants under 10 CFR part 50 and standard design certifications, combined licenses, and manufacturing licenses under 10 CFR part 52 for modifications of protection systems and safety systems, and installations of new protection system functions and safety system functions. Paragraph 50.55a(h)(3) would preserve the current licensing basis for plants in which a modification or replacement would not add new functionality, new technology, change the independence strategy, or change the diversity strategy in the existing protection system functions or safety system functions. However, licensees and applicants would be required to apply IEEE Std 603-2009 and the correction sheet dated March 10, 2015, subject to the conditions in § 50.55a(h)(4) through (8), for changes to their plant protection systems or safety systems that would add new safety functionality, new technology, or change the independence strategy or the diversity strategy in the existing protection system functions or safety system functions.

The intention of this paragraph is to assure that the most current requirements would be met for the new safety functionality or new technology being added to protection systems and safety systems. In the event the independence strategy for divisions is changed, these changes should be introduced into the protection system or safety system under the requirements in

[1] SDC – Standard design certification.

IEEE Std 603-2009 and the correction sheet dated March 10, 2015, subject to the conditions in § 50.55a(h)(4) through (8). Further, if the system diversity strategy would be changed in a protection system or safety system, the revised system diversity strategy should meet the requirements stated in IEEE Std 603-2009 and the correction sheet dated March 10, 2015, subject to the conditions in § 50.55a(h)(4) through (8), to assure the revised system diversity strategy addresses regulatory criteria.

Paragraph 50.55a(h)(3) is not intended to allow licensees to use a licensing basis or standard that results in a lessening of the requirements stated in the licensing basis for the protection system or safety system. For example, a safety system whose licensing basis meets the requirements stated in IEEE Std 603-1991 and the correction sheet dated January 30, 1995, could not be modified such that it met only the requirements stated in IEEE Std 279-1971.

The intention of providing flexible regulatory requirements is to reduce licensing uncertainty by providing consistent licensing criteria for modifications of existing protection systems and safety systems, and installations of protection system functions and safety system functions.

While the requirement in § 50.55a(h)(3) would be intended to address all cases involving modifications and installations of protection systems and safety systems, there may arise specific cases of modifications or replacements that would not apply to this paragraph. In those cases, proposed paragraph (h)(3) would require licensees and applicants to meet the requirements stated in IEEE Std 603-2009 and the correction sheet dated March 10, 2015, subject to the conditions in § 50.55a(h)(4) through (8), as this would be the most conservative approach of the alternatives for specifying protection system and safety system requirements.

The following seven examples illustrate the intended application of § 50.55a(h)(3) for different types of protection system or safety system modifications or replacements. These examples are for illustrative purposes only. These examples are summarized in Table 2.

In the first example (see Example 1 in Table 2), a licensee replaces a power supply in a single division with a new power supply that has the same functionality and technology. As part of this modification, the licensee determines that the functionality and technology of the new power supply would not be changed. The licensee determines that independence between the redundant divisions and the power trains would be maintained such that a failure occurring in the new power supply would not cause the redundant division or power train to fail. The licensee determines there would be no potential for a common cause failure to occur in the power supplies of the redundant trains. In this case, proposed § 50.55a(h)(3) would require that the protection system or safety system requirements stated in a plant's licensing basis be applicable for this modification. In modifications such as this, licensees and applicants would not be required to modify or replace an existing protection system or safety system to meet the requirements stated in IEEE Std 603-2009 and the correction sheet dated March 10, 2015, because the modification would not affect the licensing basis of the plant. A requirement to modify or replace a protection system or safety system to meet the requirements stated in IEEE Std 603-2009 and the correction sheet dated March 10, 2015, when making modifications that would not change the safety system functionality, technology (including changes to equipment qualification characteristics), independence strategy and diversity strategy could discourage licensees and applicants from improving the reliability and performance of existing protection systems, safety systems, and safety functions.

In the next example (see Example 2 in Table 2), a licensee replaces in all four divisions of the protection system pressure measurement instrumentation with new pressure measurement instrumentation that has the same function and technology (including equipment qualification characteristics). The licensee ensures the new pressure instrumentation would not change the existing independence between redundant divisions of the protection system, and the diversity strategy would not be changed. In this case, the modification would be required to meet the requirements in the license basis. A requirement to modify or replace a protection system or safety system to meet the requirements stated in IEEE Std 603-2009 and the correction sheet dated March 10, 2015, when making modifications that would not change the safety system functionality or technology could discourage licensees and applicants from improving the reliability and performance of existing protection systems, safety systems, and safety functions.

In the next example (see Example 3 in Table 2), a licensee replaces the departure from nucleate boiling ratio (DNBR) reactor trip system function with an improved DNBR reactor trip system function based on the same technology. The DNBR reactor trip system function is a diverse means of protecting the fuel rod cladding from damage caused by overheating when reactor coolant thermodynamic or thermal-hydraulic conditions (e.g., reactor coolant pressure, temperature, or coolant flow rate) become degraded such that the reactor must be shut down to prevent further overheating. This safety function is a diverse means of shutting down the reactor if the protection system fails to detect a coolant condition that could adversely affect the fuel rod cladding. The licensee determines that the proposed change would not change the safety system diversity strategy or independence between redundant divisions of the safety system. The licensee further determines that the proposed DNBR safety function would be implemented with the same system functionality. The licensee, therefore, would implement the new DNBR safety function in conformance with the plant's existing license basis instead of meeting the requirements stated in IEEE Std 603-2009 and the correction sheet dated March 10, 2015.

In the next example (see Example 4 in Table 2), a licensee modifies a microprocessor-based DNBR safety function by adding functionality to the DNBR safety function to allow the reactor operator to manually select one of four divisions of input data for each of the four previously independent DNBR divisions. This change in functionality and independence strategy would require the safety function to meet the requirements in IEEE Std 603-2009 and the correction sheet dated March 10, 2015, subject to the conditions in § 50.55a(h)(4) through (8), because the functionality and independence strategy would be changed.

In the next example (see Example 5 in Table 2), a licensee replaces an analog-based reactor protection system with a microprocessor-based reactor protection system. Proposed § 50.55a(h)(3) would require that replacement of the protection system with an equivalent protection system implemented with a different technology meet the requirements stated in IEEE Std 603-2009 and the correction sheet dated March 10, 2015, subject to the conditions in § 50.55a(h)(4) through (8). As further clarification of the intent of § 50.55a(h)(3), the new system-level functions and technology include (but are not limited to) sensor input modules, trip bistable and signal processing modules, and communication protocols for redundant divisions or external systems and trip signal voting module processors. Reusing existing components in the protection system (e.g., cables, sensors, field mounted signal conditioning equipment, control room panels, and operator displays) as a part of the system-level protection system modification would not exclude this type of modification from the requirements of IEEE Std 603-2009 and the correction sheet dated March 10, 2015, subject to the conditions in § 50.55a(h)(4) through (8). The intent of this paragraph would be to require that licensees and applicants use the most current system safety requirements available when planning, developing, and implementing new protection systems and safety systems that use functions (including changes to independence) or technology (including changes to equipment qualification characteristics) that are different from the system being replaced.

In the next example (see Example 6 in Table 2), a licensee proposes to replace a microprocessor-based DNBR safety function with another digital-based DNBR safety function. To improve availability, the licensee proposes to share all four divisions of instrument data between the DNBR safety functions, thereby reducing the independence between redundant divisions. In this example, the diversity strategy is not changed because the diversity arising from use of a DNBR function would be preserved. However, since independence between redundant divisions of the safety system would be decreased, the proposed DNBR modification would be required to meet the requirements in IEEE Std 603-2009 and the correction sheet dated March 10, 2015, subject to the conditions in § 50.55a(h)(4) through (8).

In the final example (see Example 7 in Table 2), a licensee replaces a microprocessor-based main steamline and feedwater isolation subsystem with a field-programmable gate array-based (FPGA-based) subsystem that adds new system functionality and operating characteristics that require different methods for coping with system failure modes (e.g., different common cause failure consequences that change the type of operator response and the timing of operator responses). Since system functionality and diversity strategy would be changed, the licensee would be required to meet the requirements in IEEE Std 603-2009 and the correction sheet dated March 10, 2015, subject to the conditions in § 50.55a(h)(4) through (8).

Table 2 summarizes the examples described above that illustrate the proposed § 50.55a(h) applicable requirements to be met on the basis of the scope of a modification, replacement, or installation of a protection system, safety system, or safety function. The reference to IEEE Std 603-2009 includes the IEEE Std 603-2009 correction sheet dated March 10, 2015.

Table 2 - Examples of modifications and replacements of components, functions, and systems.

Example	Modification or Replacement Example	Was Functionality, Technology, Independence strategy, or Diversity strategy changed?				Applicable Standard
		F	T	I	D	
1	Power supply replaced in one power train division.	N	N	N	N	Licensing Basis Standard
2	Pressure measurement instrumentation replaced with new pressure measurement instrumentation in all four channels of the protection system.	N	N	N	N	
3	DNBR safety function replaced with improved DNBR safety function.	N	N	N	N	
4	Added functionality to DNBR safety function to allow manual selection of one of four channels of input data for each DNBR channel.	Y	N	Y	N	IEEE Std 603-2009 (subject to the conditions in § 50.55a(h)(4) through (8))
5	Modified a protection system with components based on a different technology.	N	Y	N	N	
6	Modified channels or divisions such that independence was changed.	N	N	Y	N	
7	Modified a safety function such that protection system diversity strategy was changed.	Y	N	N	Y	

Proposed § 50.55a(h)(4) would be added to amplify the requirements stated in IEEE Std 603-2009, section 5.5, "System Integrity." Proposed § 50.55a(h)(4) would require that in order to assure the integrity and reliable operation of safety systems, safety functions shall be designed to operate in a predictable and repeatable manner. Predictable and repeatable operation of a system requires that the results of translating input signals to output signals are determined through known relationships among the controlled system states and required responses to those states, and in which a given set of input signals produce the same output signals for the full range of applicable conditions enumerated in the design basis. All signal processing between sensor data input and safety control device actuation should be accomplished in a manner such that required safety functionality remains assured regardless of responses by redundant portions of the safety system or other external systems.

Predictable and repeatable systems, in general, do not provide the capability for unscheduled event-based interrupts or operator-based system interrupts to meet system safety requirements. Systems that operate in a predictable and repeatable manner, in general, should not be designed with the capability for unscheduled event-based disruptions or operator-based system functions that would inhibit or prevent the system from meeting its safety requirements. Analyses used to demonstrate system predictability and repeatability should be based on analysis of system characteristics (e.g., definitive design and performance criteria) as opposed to probabilistic analysis.

Proposed § 50.55a(h)(5) would be added to amplify the requirements stated in IEEE Std 603-2009, section 5.6, “Independence.” Protection systems and safety systems should implement provisions for protection against identified hazards.

Proposed § 50.55a(h)(5)(i) provides requirements for applicants to address independence among redundant portions of safety systems. Receipt of information from outside a safety division may increase the likelihood of impairing the safety function in that division. Provisions should be included to protect against the potential for impairing the safety function. Redundant portions of safety systems should be sufficiently independent such that those provisions are commensurate with the relative risk posed by any potential hazards identified. The degree of interconnectivity between redundant portions of safety systems should be evaluated to ensure that the potential to introduce pathways for such hazards to propagate is minimized. Applicants should evaluate the hazards introduced by such information sharing.

Proposed § 50.55a(h)(5)(ii) provides requirements for applicants to address independence between safety systems and other systems. Receipt of information from other systems could increase the likelihood of impairing a safety function in the safety system. Provisions should be included to protect against the potential for impairing the safety function. Safety systems should be sufficiently independent from other systems such that those provisions are commensurate with the potential hazards identified. The degree of interconnectivity between safety systems and other systems should be evaluated to ensure that the potential to introduce pathways for such hazards to propagate is minimized. Applicants should evaluate the hazards introduced by such information sharing.

Section 5.6.3.1.a.2.ii and section 5.6.3.1.b in IEEE Std 603-2009 use the term "digital communications independence." This term excludes consideration of technologies other than digital that could also impair safety. Therefore, communications independence between safety systems and other systems should be applied for all signal technologies.

Proposed § 50.55a(h)(5)(iii) clarifies requirements that apply to section 5.6 of IEEE Std 603-2009. Safety system independence is a design principle that accounts for failures and interdependencies (both known and unknown) between plant systems and helps minimize the propagation of errors. To ensure independence, a safety system should not rely upon the performance or receipt of information from other external safety and/or non-safety systems to perform its safety function.

Communications independence provides a degree of protection against hazards that may impair a safety system. For example, a completely independent safety system would not have any communications link between redundant portions of safety systems or between safety and non-safety systems and therefore would be protected from the effects of communication failures or unexpected behaviors. However, having the ability to send information to non-safety systems could also be beneficial from a display, indication, diagnostic, and data recording perspective.

The sharing of signals between redundant portions of safety systems has typically only been used for the accomplishment of safety-related functions. Communications links can allow non-safety systems to be used as a means (e.g., online diagnostics) to monitor, and maintain control system parameters of a safety system. Digital technology, including the use of digital communications features may provide additional flexibility and functionality in safety and non-safety functions provided by nuclear power plant I&C systems; however, an integrated and interconnected digital communication system may also introduce additional unique failure modes and unexpected interdependencies.

Except for very simple systems, the performance of verification testing to identify all failure modes and interdependencies (e.g., latent defects) in the digital system development process is impractical, if not impossible, due to the number of input and system states that increase with the level of integration and interconnectivity. These interdependencies may challenge the independence between redundant portions of safety systems and between safety systems and non-safety systems. These failure modes and dependencies may outweigh the benefits offered by the interconnectivity.

Proposed § 50.55a(h)(5)(iii)(A) would clarify that the signal processing portions of the safety system should provide the capability to ensure that degradation or failures of signals exchanged among redundant safety divisions or between safety systems and other systems do not propagate in a manner that results in impairment of the safety functions being performed by the safety system.

Proposed § 50.55a(h)(5)(iii)(B) would clarify that safety systems should be designed with provisions for detecting and mitigating the effects of signal faults or failures received from outside the safety division. Redundant divisions of safety systems should have the capability of tolerating such faults or failures originating from outside the safety division in a manner that does not degrade the ability of the safety division to perform its safety functions.

Proposed § 50.55a(h)(5)(iii)(C) would clarify the requirements in section 5.6, “Independence” of IEEE Std 603-2009, for communications (e.g., either analog or digital signals) between redundant portions of safety systems and between safety and non-safety systems in currently operating nuclear power plant designs.

Specifically, this proposed paragraph would clarify that communications or signals received by a safety system from outside the division or system should be limited to only those that support the accomplishment of safety functions or otherwise benefit safety. Although this concept has been expressed in previous NRC guidance, the clarity of the guidance has been such that licensees and applicants have not applied this concept consistently. The safety significance of this concept warrants the need for specific regulatory criteria.

For example, complexity is increased by interconnecting safety divisions or connecting maintenance work stations to the safety system. While sharing information among redundant portions of safety systems and between safety systems and other systems could be considered a means to increase safety system reliability and performance, adding complexity to a safety system has the potential to create additional hazards that should be analyzed and addressed. Analyses should: 1) ensure the resulting system meets all the criteria in § 50.55a(h)(5), and 2) evaluate the hazards introduced by the added complexity.

Proposed § 50.55a(h)(5)(iii)(D) would clarify the requirements in section 5.6, “Independence” of IEEE Std 603-2009, for communications (e.g., either analog or digital signals) between redundant portions of safety systems and between safety and non-safety systems in new reactor designs.

Proposed § 50.55a(h)(5)(iii)(D) limits the implementation of communications between redundant portions of safety systems and between safety and non-safety systems to limit failure modes and unexpected behaviors associated with communications, while preserving the benefits of digital technology and allowing functionality that improves reliability and availability.

As a general safety principle, hazards should be eliminated when possible during the design stage; otherwise, hazards should be mitigated. Communications that use programmable means to enforce independence could introduce failure modes associated with design errors. By implementing communication independence in the hardware architectural design, the potential for the propagation of design errors is minimized. Failure modes and unexpected behaviors can be minimized in such a design by implementing redundancy in the I&C system architecture design.

Proposed § 50.55a(h)(5)(iii)(D) applies to design certifications; standard design approvals; manufacturing licenses; and combined licenses not referencing a design certification, standard design approval, or manufacturing license under 10 CFR part 52 issued on or after the effective date of this rule. Proposed § 50.55a(h)(5)(iii)(D) also applies to construction permits and operating licenses under 10 CFR part 50 issued on or after the effective date of this rule, except for an applicant for an operating license who received a construction permit for that facility before the effective date of this rule. For combined licenses issued before the effective date of the rule, § 50.55a(h)(5)(iii)(D) would only apply if the licensee modifies its data communications independence strategy.

For example, if a combined license holder modified its safety I&C system architecture by adding additional controls of safety related equipment from non-safety systems using data communications, then only the modified portion of the architecture would need to follow the applicable data communications requirements of § 50.55a(h)(5)(iii)(D) (in this example, the applicable requirement is under § 50.55a(h)(5)(iii)(D)(*3*)).

New reactors licensed under the 10 CFR part 52 process are not required to provide design implementation details at the time of design certification. As stated in § 52.47, the application must contain a level of design information sufficient to enable the Commission to reach a final conclusion on all safety questions associated with the design before the certification is granted. The requirements proposed by this rule would allow new reactors to demonstrate communications independence with a level of design information at the hardware architecture level without the need to provide detailed design implementation information, which is consistent with the requirements of § 52.47. If a new reactor applicant chooses to implement software-based solutions to enforce communications independence, additional design details and implementation information (e.g., software code, testing data, Factory Acceptance Test (FAT) results, etc.) may be needed in the licensing basis to demonstrate that the software-based solutions to enforce communications independence are safe. Based on experience of new reactor I&C systems reviews conducted prior to the development of this regulation, many applications did not have this level of information available during the time of design certification or licensing due to the state of maturity of their designs.

It is preferable from a safety and licensing point of view to design systems to promote elimination of failure modes as opposed to incorporating strategies to mitigate the results of failures. New reactor designs are able to more readily accommodate the rule as these designs do not have a current licensing basis for an existing system that may impact the particular design. However, for current reactors, this requirement does not appear to be justified from a safety standpoint. Therefore, § 50.55a(h)(5)(iii)(D) does not apply to currently operating nuclear power plant licenses or operating licenses whose construction permits were issued before the effective date of the rule.

The proposed independence requirements would increase consistency of the regulatory framework for I&C systems with the Commission's policy on advanced reactors by having a simplified means to accomplish safety functions. This approach is supported by the 2007 National Academy of Science Study, "Software for Dependable Systems: Sufficient Evidence?"

(National Research Council. *Software for Dependable Systems: Sufficient Evidence?* Washington, DC: The National Academies Press, 2007), which linked the issue of complexity to the independence design principle. Specifically, the study noted that "the most important form of simplicity is that produced by independence, in which particular system-level properties are guaranteed by individual components much smaller than the system as a whole, which can preserve these properties despite failures in the rest of the system. Independence can be established in the overall design of the system, with the support of architectural mechanisms."

Non-safety digital I&C systems could have failure modes and behaviors for which a complete set of failure modes may not be fully identified or adequately mitigated. Specifically, since non-safety systems may not have been developed using rigorous development activities (e.g., independent V&V, requirements traceability) that are required for safety systems, there is more potential for the software in these non-safety systems to contain errors and defects. It is this potential for latent software design errors and/or hardware defects that may create failure modes and/or unexpected behavior within the non-safety system that may propagate to safety systems through the communications links of interconnected systems. Proposed § 50.55a(h)(5)(iii)(D)(*1*) would eliminate or mitigate failure modes and unexpected behaviors associated with communication failures among interconnected I&C systems by restricting use of communication links from non-safety systems to safety systems during specific periods of operation.

A further concern regarding non-safety systems is that they are not required to operate in a predictable and repeatable manner (e.g., no response time requirements, using event-driven interrupts). This situation could potentially increase or introduce unidentified failure modes within these non-safety systems. Although safety-related isolation devices can be used to detect and prevent propagation of failures from non-safety systems to safety systems, these isolation devices may not be capable of addressing the effects of failures originating in non-safety systems because the full set of non-safety system failure modes may not be identified or anticipated. In addition, a safety system's ability to address potential failures (e.g., communications errors) propagated by non-safety systems may not be effective in addressing these failures. This situation may arise when the potential failures occur in a manner different than anticipated, and thus the software features in the safety system may not be able to detect or mitigate an unanticipated failure.

Proposed § 50.55a(h)(5)(iii)(D)(*1*) is intended to ensure that data communication from safety systems to non-safety systems is in one direction while the safety system division or channel is in operation, and the one-way communication is accomplished through hardware means. This will allow information to be transmitted to non-safety systems in a manner that prevents the receiving non-safety system from adversely impacting a safety function. By limiting the implementation of the data communication to one direction from the safety system to the non-safety system while the safety system division or channel is in operation, this paragraph allows for safety and non-safety systems to take advantage of digital technology without adversely affecting safety system functionality.

For example, the proposed paragraph allows communication from safety systems to non-safety systems for display, control, recording, and diagnostics. Failure modes may still exist with use of data communications within the design; however, if the communication link is a physical one-way connection (i.e., no hand-shaking signal and only a fiber optic or copper wire connection from a transmit port to a receive port), then the failure modes associated with data communications are effectively addressed by hardware designed to maintain the communication flow. The use of physical means (e.g., hardware devices) to prevent non-safety to safety system communication while the safety system division or channel is in operation further reduces reliance on software to maintain safety system independence.

Proposed § 50.55a(h)(5)(iii)(D)(_2_) ensures that transfer of signals between redundant portions of safety systems is only accomplished when the signal transferred is required for the performance of safety-related functions. Although sharing of signals among redundant portions of safety systems could be considered a means to increase safety system reliability, operational performance, and availability, such sharing of signals has the potential to create additional failure modes and unexpected behaviors. The NRC recognizes that there may exist circumstances in which the sharing of information is necessary to accomplish a safety function. The sharing of inputs to the coincidence logic (i.e., combining the logical results of each division to produce a safety system actuation signal) among otherwise independent redundant portions of the protection system has been found acceptable when this communication is required to accomplish safety-related functions or to perform safety interlock functions.

Proposed § 50.55a(h)(5)(iii)(D)(_3_) ensures that, for functions that require safety systems to receive signals from non-safety systems to ensure diversity and defense-in-depth or to support automatic anticipatory reactor trip functions, the signal transfer method is restricted to means that do not use data communication. For example, diverse back-up systems may require connection to safety components to mitigate the effects of beyond design basis safety system common-cause failures. If the diverse back-up system is a non-safety system, then functionality of this system would be limited to mitigating the effects of beyond design basis safety system common-cause failures (e.g., the non-safety system should not have the capability to perform control functions or modify safety-related functions during normal operations). Another example is a nuclear power plant design that implements anticipatory reactor trip functions (e.g., reactor shutdown on turbine trip). In these cases, a signal may need to be sent from a non-safety system to the reactor protection system to initiate the anticipatory reactor trip function.

If a signal is needed to support diversity or automatic anticipatory reactor trip functions as described in the examples above, then independence would be achieved through means other than data communications. These alternative means could be accomplished using Class 1E isolators. As required by § 50.55a(h)(5)(ii), the hazards associated with the transmission of these signals over hardwired connections (e.g., EMI, spurious actuations) are to be identified and addressed such that it can be demonstrated that a fault in the non-safety system would not propagate to the safety system. The above requirements limit the transfer of signals from non-safety systems to safety systems to reduce interdependencies between safety systems and non-safety systems.

Proposed § 50.55a(h)(5)(iii)(D)(3) limits transmission of signals to safety systems from other systems to only those that are necessary to accomplish defense-in-depth, diversity, or automatic anticipatory reactor trip functions. This paragraph does not allow for control of safety equipment from non-safety systems (e.g., non-safety control systems and a multi-divisional display for controlling safety systems). In addition to the potential for errors in non-safety systems to impact the operation of safety systems, control of plant safety equipment could result in conditions that exceed a plant's safety analysis limits. For example, failures in non-safety systems might result in spurious actuation of safety systems that result in plant conditions that exceed safety analysis limits. Limiting the control of safety equipment from non-safety systems reduces the potential for such spurious actuations.

Proposed § 50.55a(h)(5)(iii)(D)(4) addresses the potential communication pathways introduced by an alternative approach to § 50.55a(h) between a digital safety system and other systems, such as other safety systems or non-safety systems. This paragraph would require applicants of design certifications, standard design approvals, or manufacturing licenses to identify all direct and indirect communication pathways to safety systems to facilitate the identification of interdependences and failure modes in the design. For example, if a non-safety system is connected to a safety system (e.g., either directly connected or indirectly through another non-safety system) to provide information on the status of the plant, then this connection would need to be identified to ensure that failure modes and unexpected behaviors associated with this connection are addressed.

Proposed § 50.55a(h)(6) would be added to correct a reference in IEEE Std 603-2009 section 6.5.1, "Checking the operational availability."

Section 6.5.1.b in IEEE Std 603-2009 references section 6.6, "Operating Bypasses." Section 6.6 requires safety systems to automatically override a safety function bypass condition when plant operating conditions require the safety function to be active, which is not relevant to checking operational availability. Section 6.7, "Maintenance Bypass," requires safety systems to accomplish safety functions while sense and command features equipment is in maintenance bypass, which is relevant to checking operational availability. Since section 6.5.1 addresses checking operational availability of safety functions, which is a maintenance activity, licensees should reference IEEE Std 603-2009 section 6.7, which addresses system bypasses during maintenance activities instead of referencing section 6.6.

Proposed § 50.55a(h)(7) would clarify requirements with regard to the ability of the safety system to continue to perform its required safety functions while redundant portions are in maintenance bypass mode. The paragraph also clarifies the need to demonstrate acceptable reliability of the portions of the safety system that are not in maintenance bypass mode.

Section 6.7 in IEEE Std 603-2009 states:

> Capability of a safety system to accomplish its safety function shall be retained while sense and command features equipment is in maintenance bypass. During such operation, the sense and command features should continue to meet the requirements of 5.1 and 6.3.
>
> NOTE—For portions of the sense and command features that cannot meet the requirements of 5.1 and 6.3 when in maintenance bypass, acceptable reliability of equipment operation shall be demonstrated (e.g., that the period allowed for removal from service for maintenance bypass is sufficiently short, or additional

> measures are taken, or both, to ensure there is no significant detrimental effect on overall sense and command feature availability).

In IEEE standards, notes provide additional information concerning a particular requirement and do not provide mandatory requirements. A "NOTE" in the text of a requirement in an IEEE standard is an informative (i.e., non-binding) part of the standard; therefore, the IEEE does not allow important information on safety, health, or the environment in a note. Therefore, the note in IEEE Std 603-2009 section 6.7 would not become a regulatory requirement or alternative to the requirement(s) in the referencing section although the IEEE Std 603-2009 would be incorporated by reference in § 50.55a.

In contrast, section 6.7 in IEEE Std 603-1991 states:

> Capability of a safety system to accomplish its safety function shall be retained while sense and command features equipment is in maintenance bypass. During such operation, the sense and command features shall continue to meet the requirements of [section] 5.1 and [section] 6.3.
>
> EXCEPTION: One-out-of-two portions of the sense and command features are not required to meet [section] 5.1 and [section] 6.3 when one portion is rendered inoperable, provided that acceptable reliability of equipment operation is otherwise demonstrated (that is, that the period allowed for removal from service for maintenance bypass is sufficiently short to have no significantly detrimental effect on overall sense and command features availability).

Section 6.7 in IEEE Std 603-1991, as compared to section 6.7 in IEEE Std 603-2009, provides a more conservative requirement for placing sense and command features equipment in maintenance bypass. Therefore, proposed § 50.55a(h)(7) would require that licensees and applicants meet the requirements stated in section 6.7 of IEEE Std 603-1991.

Proposed § 50.55a(h)(8) would provide a requirement that applicants and licensees develop and maintain documentation, analyses, and design details demonstrating compliance with § 50.55a(h)(2) through (7) of this section. The NRC intends that this documentation be accessible to the NRC staff to support independent NRC evaluations of safety systems.

As will be discussed in section XIV, "Backfitting and Issue Finality," of this document, the proposed rule would apply to the Watts Bar Nuclear Plant, Unit 2, and the Bellefonte Nuclear

Plant, Units 1 and 2, but only if the construction permit holder makes changes or modifications to, or replaces, the plant's safety system or protection system (as reviewed and approved in the construction permit application and described in the preliminary safety analysis reports) under § 50.55a(h)(3) of the proposed rule. There are several reasons for this determination. First, on July 25, 2007, the Commission approved the NRC staff's recommendation that the licensing basis for Watts Bar Nuclear Plant, Unit 1, serve as the licensing basis for the review and licensing of Watts Bar Nuclear Plant, Unit 2. This means that Watts Bar Nuclear Plant, Unit 2, would receive the same regulatory treatment as the currently operating Watts Bar Nuclear Plant, Unit 1 (see Staff Requirements Memorandum, SECY-07-0096 – Possible Reactivation of Construction and Licensing Activities for the Watts Bar Nuclear Plant, Unit 2, July 25, 2007, ADAMS Accession No. ML072060688). Second, this approved staff recommendation is included in the NRC staff's plan "to implement the Commission Policy Statement on Deferred Plants in its review of the WBN [Watts Bar Nuclear Unit] 2 reactivation and OL [operating license] application" (see SECY-07-0096, "Possible Reactivation of Construction and Licensing Activities for the Watts Bar Nuclear Plant Unit 2," June 7, 2007, ADAMS Accession No. ML071220492). The previously mentioned Commission Policy Statement on Deferred Plants (52 FR 38077; October 14, 1987) states that "[d]eferred plants of custom or standard design will be considered in the same manner as plants still under construction with respect to applicability of new regulations, guidance, and policies." Therefore, because § 50.55a(h)(2)(ii), would allow nuclear power plants with construction permits issued after January 1, 1971, but before May 13, 1999, to use IEEE Std 279-1968, IEEE Std 279-1971, IEEE Std 603-1991 and the correction sheet dated January 30, 1995, or IEEE Std 603-2009 and the correction sheet dated March 10, 2015, this proposed rule would apply to Watts Bar Nuclear Plant, Unit 2 (construction permit issued in 1973), and Bellefonte Nuclear Plant, Units 1 & 2 (construction permits issued in

1974), only if the proposed changes, modifications, or replacements are initiated by the construction permit holder.

Paragraph 50.69(b)(1)(v), “Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors,” would be changed to add references to section 5.3 and section 5.4 in IEEE Std 603-2009. These changes would update § 50.69 to reference the applicable sections from IEEE Std 603-2009, in addition to the corresponding sections in IEEE Std 279 and IEEE Std 603-1991 that are already referenced in § 50.69.

In 10 CFR part 50, appendix E, “Emergency Planning and Preparedness for Production and Utilization Facilities,” section VI, “Emergency Response Data Systems,” subsection 2.a, references footnote 7. This footnote would be changed to reference the correct title of § 50.55a(h) as “Protection and safety systems.” This would be an administrative change that would not affect the existing appendix E requirements.

The following paragraphs describe regulatory considerations associated with the requirements stated in IEEE Std 603-2009 and the correction sheet dated March 10, 2015.

1. The IEEE Std 603-2009 references several industry codes and standards. These referenced standards are not within the scope of this rule, are not approved for incorporation by reference, and are not approved by the NRC by this rulemaking. These referenced standards are not mandatory NRC requirements (even though IEEE Std 603-2009 invokes the referenced standards by the use of “shall”). If a referenced standard has been endorsed in a regulatory guide, the standard constitutes a method acceptable to the NRC for meeting a regulatory requirement. In many cases, a regulatory guide endorses a previous version of an IEEE standard. These guides represent current NRC recommended practices. Licensees and applicants may opt to use alternate approaches to meet the requirements stated in § 50.55a(h) if the licensee or applicant can provide a sufficient technical basis for the alternate approach.

2. In section 4.g, the IEEE Std 603-2009 includes electromagnetic interference as an additional environmental factor in the design basis. The NRC agrees that electromagnetic interference should be part of the environmental factors in the design basis. The NRC guidance on this subject is provided in RG 1.180, Revision 1, "Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems," dated October 2003 (ADAMS Accession No. ML032740277).

3. In section 5.1, IEEE Std 603-2009 states that a single failure could occur prior to, or at any time during a design basis event for which the safety system is required to function. This clarification is consistent with the NRC position and was determined to be acceptable. The NRC guidance on this subject is provided in RG 1.53, Revision 2, "Application of the Single-Failure Criterion to Safety Systems," dated November 2003 (ADAMS Accession No. ML033220006).

4. Section 5.4 in IEEE Std 603-2009 references IEEE Std 323-2003, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations," as this standard is the latest version of the equipment qualification standard. The IEEE Std 323-2003 is endorsed by RG 1.209, "Guidelines for Environmental Qualification of Safety-Related Computer-Based Instrumentation and Control Systems in Nuclear Power Plants," dated March 2007 (ADAMS Accession No. ML070190294) for providing criteria for computer-based equipment qualification in mild environments. The NRC does not endorse IEEE Std 323-2003 as an acceptable means of meeting regulatory requirements for qualifying equipment for operations in harsh environments. For equipment qualified for harsh environments, the procedures described by IEEE Std 323-1974, "IEEE Standard for Qualifying Class IE Equipment for Nuclear Power Generating Stations," are acceptable to the NRC staff for satisfying the NRC's regulations pertaining to the qualification of electric equipment for service in nuclear power plants to ensure that the equipment can perform its

safety functions in harsh environments subject to the regulatory positions described in RG 1.89, Revision 1, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants," dated June 1984 (ADAMS Accession No. ML003740271).

Staff finds using two RGs to endorse the same IEEE standard to be appropriate because RG 1.209 applies to computer-based equipment operating in mild environments and RG 1.89 applies to equipment operating in harsh environments. The guidance in RG 1.209 (endorsing IEEE Std 323-2003) complements the guidance in RG 1.89 (endorsing IEEE Std 323-1974), which was not changed because the new version of IEEE Std 323-2003 did not change any of the criteria applicable to equipment under the scope of § 50.49. Therefore, it is appropriate to reference IEEE Std 323-1974 via RG 1.89 for qualifying equipment operating in harsh environments.

5. Section 5.16, "Common-cause failure criteria," of IEEE Std 603-2009 does not provide specific guidance for performing an engineering evaluation of common-cause failures (CCF); instead this section states that IEEE Std 7-4.3.2-2003 provides guidance on this topic. As discussed previously, this standard is not approved for incorporation by reference and is not approved by the NRC by this rulemaking. The use of digital technology in safety systems has led to concerns that errors could lead to CCFs that might disable one or more safety functions in redundant divisions of a safety system. Errors can be introduced into a system at any stage of the system development life cycle, including specification, development of requirements, design, implementation, integration, maintenance, or modification. A fault is systemic if it exists in multiple components in an integrated instrumentation and control system. A systemic fault becomes a CCF if a triggering event occurs that causes concurrent failures in multiple divisions of the safety system, thereby defeating one or more safety functions. Safety systems must have adequate diversity and defense-in-depth to compensate for CCFs. Digital safety system CCFs generally are not subject to the single failure criteria of IEEE Std 379-2000; however,

software CCFs are required to be addressed by performing a diversity and defense-in-depth analysis as part of meeting the requirements of GDC 22. In performing a diversity and defense-in-depth analysis, the applicant or licensee should analyze each postulated CCF for each event that is evaluated in the safety analysis report (SAR) section analyzing power operation accidents at the plant conditions corresponding to the event. This analysis may use best-estimate assumptions (i.e., realistic assumptions) to analyze the plant response to design-basis events, or the conservative assumptions on which the Chapter 15 SAR analysis is based. The conditions under which a postulated software CCF concurrent with events evaluated in the accident analysis section of the SAR are considered beyond-design-basis conditions. Consequently, the diversity and defense-in-depth analysis may credit non-safety systems in the analysis if the non-safety system is of sufficient quality to perform the necessary function under the postulated event conditions. Branch Technical Position 7-19, "Guidance for Evaluation of Diversity and Defense-in-Depth in Digital Computer-Based Instrumentation and Control Systems," in NUREG-0800, "Standard Review Plan," Chapter 7, "Instrumentation and Controls," provides guidance for evaluating and mitigating software CCFs.

6. Section 6.5.1.b references section 6.6, "Operating Bypasses," in IEEE Std 603-2009. Section 6.6 requires safety systems to automatically override a safety function bypass condition when plant operating conditions require the safety function to be active. Section 6.7, "Maintenance Bypass," requires safety systems to accomplish safety functions while sense and command features equipment is in maintenance bypass. Section 6.5.1 should reference section 6.7 instead of section 6.6 because section 6.7 addresses maintenance activities performed while a reactor is in operation, whereas, section 6.6 addresses overriding bypasses. In order to maintain consistency with the subject of section 6.5.1.b and the requirement for maintaining safety system functionality during maintenance operations while a reactor is in

operation, licensees should reference IEEE Std 603-2009 section 6.7, which addresses system bypasses during maintenance activities.

7. Section 6.7 of IEEE Std 603-2009 states that the capability of a safety system to accomplish its safety function shall be retained while sense and command features equipment is in maintenance bypass. During such operation, the sense and command features should continue to meet the single failure criterion of section 5.1 and the interaction between the sense and command features and other systems criterion of section 6.3. In order to maintain consistency with GDC 21, "*Protection system reliability and testability,*" the NRC staff would incorporate regulatory requirements in § 50.55a(h)(7) of the rule to maintain the current regulatory requirements. Draft Regulatory Guide 1251, "Criteria for the Power, Instrumentation, and Control Portions of Safety Systems for Nuclear Power Plants," would provide additional guidance for implementing the requirements of the rule.

8. The criteria proposed by this rule would delineate when the current licensing basis could be used for modifications or replacements of protection systems and safety systems, and when these modifications and replacements would be required to meet the requirements in IEEE Std 603-2009 and the correction sheet dated March 10, 2015, subject to the conditions in § 50.55a(h)(4) through (8). The rule is not intended to require that these system modifications or replacements be submitted to the NRC for approval prior to implementing a plant change. Modifications, additions to, or removal of protection or safety system safety functions from a licensed facility that affect the design function would be submitted to the NRC for approval in accordance with § 50.59, "Changes, tests, and experiments." Changes to a licensed facility would continue to be reported to the NRC in accordance with § 50.71, "Maintenance of records, making of reports."

B. Conforming Changes.

The proposed rule contains conforming changes to 10 CFR 50.69(b)(1)(v) to add references to section 5.3 and section 5.4 in IEEE Std 603-2009.

The proposed rule also contains conforming changes to 10 CFR part 50, appendix E, “Emergency Planning and Preparedness for Production and Utilization Facilities,” footnote 7 to change the referenced heading of paragraph 50.55a(h) from “Protection systems” to the proposed new heading for paragraph 50.55a(h), “Protection and safety systems.”

C. Non-concurrences.

Several NRC staff individuals did not agree with some provisions of the proposed rule and submitted four non-concurrences on the proposed rule. In accordance with the NRC’s non-concurrence process, NRC staff management assessed the non-concurrence issues and revised the proposed rule to address some of the staff individuals’ reasons for their non-concurrence. Despite these changes, the staff members reaffirmed their non-concurrence. The four non-concurrences can be found at ADAMS Accession Nos. ML14280A340, ML14280A367, ML14281A145, and ML15036A467. The non-concurrences include the NRC staff management resolution of the non-concurrences and descriptions of the changes to the proposed rule directed by management.

IV. Section-by-Section Analysis.

Paragraph 50.55a(a)(2)(iii)

This paragraph would be revised to remove the reference to § 50.55a(h)(3).

Paragraph 50.55a(a)(2)(iv)

This paragraph would be revised to remove the reference to § 50.55a(h)(3).

Paragraph 50.55a(a)(2)(v)

This paragraph would be added to include IEEE Std 603-2009, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations."

Paragraph 50.55a(a)(2)(vi)

This paragraph would be added to include IEEE Std 603-2009 correction sheet dated March 10, 2015, "Errata to IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations."

Paragraph 50.55a(h)(1)

This paragraph would be revised to include definitions for the terms "current reactors" and "new reactors" in the context of § 50.55a(h).

Paragraph 50.55a(h)(2)(i)

This paragraph would be added to clarify the requirements for protection systems and safety systems in nuclear power plants with construction permits issued before January 1, 1971. Licensees of plants in this category would not be required to change the licensing basis of their plants to meet the requirements stated in IEEE Std 603-2009 and the correction sheet dated March 10, 2015, for the protection systems and safety systems that are not subject to the requirements stated in § 50.55a(h)(3). Optionally, licensees would be allowed to change the licensing basis of their plants to meet the requirements stated in IEEE Std 603-2009 and the correction sheet dated March 10, 2015, subject to the conditions in § 50.55a(h)(4) through (8).

Paragraph 50.55a(h)(2)(ii)

This paragraph would be added to clarify the requirements for protection systems and safety systems in nuclear power plants with construction permits, standard design certifications issued on or after January 1, 1971, but before May 13, 1999. Protection systems and safety systems that are not subject to the requirements of § 50.55a(h)(3) would be required to meet the requirements stated in the plant's licensing basis or in the standard design certification rule or standard design approval after the effective date of this rule instead of the requirements stated in IEEE Std 603-2009 and the correction sheet dated March 10, 2015. Applicants and licensees would have the option to meet the requirements stated in IEEE Std 603-2009 and the correction sheet dated March 10, 2015, subject to the conditions in § 50.55a(h)(4) through (8), for the protection systems or safety systems.

Paragraph 50.55a(h)(2)(iii)

This paragraph would be added to clarify the requirements for protection systems and safety systems in standard design certifications issued before May 13, 1999. These protection systems and safety systems would be required to meet the requirements stated in the plant's licensing basis after the effective date of this rule instead of the requirements stated in IEEE Std 603-2009 and the correction sheet dated March 10, 2015, regardless of the date the COL is issued. Licensees of plants in this category would not be required to change the licensing basis of their plants to meet the requirements stated in IEEE Std 603-2009 and the correction sheet dated March 10, 2015, for the protection systems and safety systems that are not subject to the requirements stated in § 50.55a(h)(3). Applicants and licensees also would have the option to meet the requirements stated in IEEE Std 603-2009 and the correction sheet dated March 10, 2015, subject to the conditions in § 50.55a(h)(4) through (8), for the protection systems or safety systems.

Paragraph 50.55a(h)(2)(iv)

This paragraph would be added to clarify the requirements for protection systems and safety systems in standard design certifications issued on or after May 13, 1999, but before the effective date of this rule. The protection systems and safety systems that are not subject to the requirements stated in § 50.55a(h)(3) would be required to meet the requirements stated in the standard design certification licensing basis after the effective date of this rule instead of the requirements stated in IEEE Std 603-2009 and the correction sheet dated March 10, 2015, regardless of the date a construction permit is issued for the standard design certification. Applicants and licensees also would have the option to meet the requirements stated in IEEE Std 603-2009 and the correction sheet dated March 10, 2015, subject to the conditions in § 50.55a(h)(4) through (8), for the protection systems or safety systems.

Paragraph 50.55a(h)(2)(v)

This paragraph would be added to require that protection systems and safety systems for nuclear power plants that reference standard design certifications, and combined licenses issued after the effective date of this rule meet the requirements stated in IEEE Std 603-2009 and the correction sheet dated March 10, 2015, subject to the conditions in § 50.55a(h)(4) through (8).

Paragraph 50.55a(h)(2)(vi)

This paragraph would be added to require that protection systems and safety systems in construction permits under 10 CFR part 50 for applications submitted to the NRC after the effective date of this rule would be required to meet the requirements stated in IEEE Std 603-2009 and the correction sheet dated March 10, 2015, subject to the conditions in § 50.55a(h)(4) through (8). Construction permits issued under 10 CFR part 50 that are issued before the effective date of this rule would continue to be required to meet the requirements in their licensing bases.

Paragraph 50.55a(h)(2)(vii)

This paragraph would be added to require that combined licenses and manufacturing licenses that reference a standard design certification issued before the effective date of this rule meet the requirements stated in the referenced standard design certification. Safety system designs in combined licenses and manufacturing licenses that reference a standard design certification issued after the effective date of this rule would be required to meet the requirements stated in IEEE Std 603-2009 and the correction sheet dated March 10, 2015, subject to the conditions in § 50.55a(h)(4) through (8).

Paragraph 50.55a(h)(3)

This paragraph would be added to require that licensees meet the requirements in the nuclear facility current licensing basis standard or the requirements in IEEE Std 603-2009 and the correction sheet dated March 10, 2015, (subject to the conditions stated in § 50.55a(h)(4) through (8)) when performing modifications to the protection system or safety system. The purpose of this paragraph is to specify the licensing basis standard requirements on the basis of the scope of modifications being performed on a protection system or safety system. Modifications that would not change the functionality of a safety system, the underlying technology of a safety system, division independence strategy in a safety system, or the diversity strategy in a safety system would be required to conform to the facility's current licensing basis standard. All other changes would be required to meet the requirements in IEEE Std 603-2009 and the correction sheet dated March 10, 2015, subject to the conditions in § 50.55a(h)(4) through (8).

Paragraph 50.55a(h)(4)

This paragraph would be added to amplify the requirements in IEEE Std 603-2009 and the correction sheet dated March 10, 2015, regarding the system integrity.

Paragraph 50.55a(h)(5)

This paragraph would be added to amplify the requirements in IEEE Std 603-2009 and the correction sheet dated March 10, 2015, regarding independence between safety divisions and between safety systems and non-safety systems.

Paragraph 50.55a(h)(6)

This paragraph would be added to maintain consistency with the subject of section 6.5.1.b and the requirement for maintaining safety system functionality during maintenance operations while a reactor is in operation by referencing section 6.7, “Maintenance Bypass” instead of section 6.6, “Operating Bypasses.”

Paragraph 50.55a(h)(7)

This paragraph would be added to amplify the requirements in IEEE Std 603-2009 section 6.7, “Maintenance Bypass.”

Paragraph 50.55a(h)(8)

This paragraph would be added to require licensees and applicants to maintain documents and records that demonstrate compliance with § 50.55a(h)(2) through (7) of this section.

Paragraph 50.69(b)(1)(v)

This paragraph would be amended to include a reference to sections 5.3 and 5.4 in IEEE Std 603-2009 as a result of incorporating by reference IEEE Std 603-2009 in paragraph 50.55a(a).

10 CFR part 50, appendix E, footnote 7

This footnote would be amended to change the existing title in the citation of § 50.55a(h) from "*Protection Systems*" to "*Protection and Safety Systems*."

V. Specific Request for Comments.

The NRC requests public comment on the changes proposed by this rule and on the regulatory analysis as discussed in section IV, "Section-by-Section Analysis," and in section XIII, "Regulatory Analysis: Availability." In addition, the NRC requests public comment on the associated draft regulatory guide for this rule. The NRC also requests stakeholders to consider specific questions regarding the process to implement changes to § 50.55a(h). The NRC staff would like input on ways to make future rulemakings that incorporate by reference future versions of IEEE Std 603 into § 50.55a(h) more predictable and consistent.

The IEEE standards are subject to review at least every 10 years, for revision or removal. When an IEEE standard is more than 10 years old and has not been revised by the IEEE, it is reasonable to conclude that its contents, although still of some value, do not wholly reflect advances in technology or issues arising from technologies that had been developed or introduced into the nuclear industry since publication of the previous standard. It is, therefore, appropriate to periodically consider for rulemaking the content of newer versions of an IEEE standard.

The requirements stated in IEEE Std 603-2009 and the correction sheet dated March 10, 2015, are, for some topics, different from the requirements contained in previous versions of this standard. Consequently, licensees may be required to adhere to different requirements when they modify protection systems or safety systems over an extended period (e.g., over several refueling outages). The point at which IEEE Std 603-2009 and the correction sheet dated March 10, 2015, would become the underlying basis for the modified protection system or safety system is not specifically addressed in this proposed rule.

The IEEE Std 603-2009 references IEEE Std 7-4.3.2-2003, “IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations,” and indicates this document is indispensable to the implementation of IEEE Std 603-2009. The IEEE Std 7-4.3.2-2003 further states that the criteria contained within that document, in conjunction with criteria of IEEE Std 603, establishes minimal functional and design requirements for computers used as components of a safety system.

Also, some of the changes proposed by this rule will change the current NRC requirements and practices associated with independence and other aspects of safety system design.

Accordingly, the NRC is requesting responses to the following questions:

1. How frequently should the NRC conduct rulemaking to incorporate by reference the IEEE Std 603 into § 50.55a(h)?

2. What would be a reasonable compliance period for applications or license amendments? For example, should the NRC allow 6 months after publication of a final rule amending § 50.55a(h) before license applications or amendments submitted to the NRC be required to follow the new requirements?

3. Licensees could replace protection systems or safety systems using new functionality or technology over an extended period (e.g., over several refueling outages). At what point in

this extended period of modification should the NRC require the protection system or safety system to meet the requirements stated in IEEE Std 603-2009 and the correction sheet dated March 10, 2015? Further, should the NRC also require the parts of the protection system or safety system that were added or modified up to that point to meet the requirements stated in IEEE Std 603-2009 and the correction sheet dated March 10, 2015?

4. Will the proposed independence requirements (§ 50.55a(h)(5)) provide more regulatory certainty for new and current reactor I&C designs? Are there better regulatory criteria to achieve independence than those being proposed? What additional guidance is necessary to implement the proposed criteria?

5. How likely is it that applicants and licensees will use the alternative process (as provided in § 50.55a(z)) associated with the new requirements for "independence" (IEEE Std 603-2009, section 5.6)? In what respects would alternatives be sought and what would be the basis for seeking the alternatives?

6. Will the proposed rule language act to limit different design solutions to address independence? If yes, what is the net impact on plant safety?

7. Will the added requirements and restrictions on digital communications independence discourage the nuclear industry from using available technologies to enhance safety system performance or replace aging and obsolete safety systems?

8. Will different requirements for digital system independence for new and current reactors lead to inconsistencies between reactor designs that will impact safety or the ability of the NRC to effectively carry out inspections or regulatory reviews?

9. IEEE Std 603-2009, Clause 5.16, "Common-cause failure criteria," does not provide specific requirements for addressing common-cause failure and the proposed rule does not provide requirements in this area. Should the NRC provide requirements within the final rule addressing common-cause failure criteria?

10. The Commission provided defense-in-depth and diversity criteria to address potential common-cause failures in the Staff Requirements Memorandum to SECY-93-087. These criteria are used by the staff in their licensing reviews in accordance with Branch Technical Position 7-19, "Guidance for Evaluation of Diversity and Defense-in-Depth in Digital Computer-Based Instrumentation and Control Systems," of NUREG-0800, "Standard Review Plan." Should these criteria be included in this rulemaking, or should other criteria be included?

11. Given that (1) the Staff Requirements Memorandum to SECY-93-087 was originally written to address advanced reactors (i.e., design certifications under review at that time); (2) new and operating reactors face different I&C challenges such as analog-to-digital upgrades; and (3) defense-in-depth and diversity analyses can promote better understanding, particularly for new and first-of-a-kind reactor designs having little to no operating history, if the common-cause failure criteria is included in the rule, should it be applicable to new reactors only?

The NRC will review the responses to these questions to help determine agency positions on the scope, frequency, and methods to communicate the incorporation by reference of IEEE Std 603 rulemakings.

VI. Request for Comment: Draft Regulatory Guide.

Draft Regulatory Guide (DG) – 1251, "Criteria for the Power, Instrumentation, and Control Portions of Safety Systems for Nuclear Power Plants," (Regulatory Guide 1.153, Revision 2; ADAMS Accession No. ML112160394) would provide additional guidance for implementing the requirements of the rule. The DG-1251 is based upon the discussion in this FRN and does not modify the scope of paragraph 50.55a(h). The NRC requests public comment on the draft regulatory guide.

VII. Plain Writing.

The Plain Writing Act of 2010 (Pub. L. 111-274) requires Federal agencies to write documents in a clear, concise, and well-organized manner. The NRC has written this document to be consistent with the Plain Writing Act as well as the Presidential Memorandum, “Plain Language in Government Writing,” published June 10, 1998 (63 FR 31883). The NRC requests comment on this document with respect to the clarity and effectiveness of the language used.

VIII. Availability of Documents.

The NRC is making the documents identified in Table 3 available to interested persons through one or more of the following methods, as indicated. To access documents related to this action, see the ADDRESSES section of this document.

Table 3 - Documents referenced in proposed 10 CFR 50.55a rulemaking.

Document	ADAMS Accession No. / Web site
Proposed Rule Documents	
SECY-15-XXXX, “Proposed Rule: Incorporation by Reference of Institute of Electrical and Electronics Engineers Standard 603-2009, ‘IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations’ (RIN 3150-AI98).”	ML113191143
Staff Requirements Memorandum for SECY-15-XXXX, “Proposed Rule: Incorporation by Reference of Institute of Electrical and Electronics Engineers Standard 603-2009, ‘IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations’ (RIN 3150-AI98).”	(Not available. Will be inserted before publication of FRN.)
DG-1251 (RG 1.153, Rev. 2), “Criteria for the Power, Instrumentation, and Control Portions of Safety Systems for Nuclear Power Plants.”	ML112160394
Draft Regulatory Analysis for Proposed Rulemaking: “Incorporation by Reference of Institute of Electrical and Electronics Engineers Standard 603-2009.”	ML120310194
Non-Concurrence on Proposed Rule to Incorporate by Reference Institute of Electrical and Electronics Engineers	ML14280A340

Standard 603-2009 (NCP-2014-001).	
Non-Concurrence on Proposed Rule to Incorporate by Reference Institute of Electrical and Electronics Engineers Standard 603-2009 (NCP-2014-003).	ML14280A367
Non-Concurrence on Proposed Rule to Incorporate by Reference Institute of Electrical and Electronics Engineers Standard 603-2009 (NCP-2014-004).	ML14281A145
Non-Concurrence on Proposed Rule to Incorporate by Reference Institute of Electrical and Electronics Engineers Standard 603-2009 (NCP-2015-001).	ML15036A467
NRC Guidance and Technical Documents	
Branch Technical Position 7-19, "Guidance for Evaluation of Diversity and Defense-in-Depth in Digital Computer-Based Instrumentation and Control Systems."	ML110550791
NUREG-0800, Chapter 7, Section 7.0, Rev. 6, "Instrumentation and Controls – Overview of Review Process."	ML100740146
NUREG/CR-6303, "Method for Performing Diversity and Defense-in-Depth Analyses of Reactor Protection Systems."	ML071790509
RG 1.53, Rev. 2, "Application of the Single-Failure Criterion to Safety Systems."	ML033220006
RG 1.89, Rev. 1, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants."	ML003740271
RG 1.152, Rev. 3, "Criteria for Use of Computers in Safety Systems of Nuclear Power Plants."	ML102870022
RG 1.153, Rev. 1, "Criteria for Safety Systems."	ML003740022
RG 1.180, Rev. 1, "Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems."	ML032740277
RG 1.209, "Guidelines for Environmental Qualification of Safety-Related Computer-Based Instrumentation and Control Systems in Nuclear Power Plants."	ML070190294
SECY-07-0096, "Possible Reactivation of Construction and Licensing Activities for the Watts Bar Nuclear Plant Unit 2."	ML071220492
Staff Requirements – SECY-93-087 – Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs	ML003708056
Staff Requirements – Affirmation Session, 11:30 A.M., Friday, September 10, 1999, Commissioners' Conference Room, One White Flint North, Rockville, Maryland (Open To Public Attendance).	ML003751061
Staff Requirements – SECY-07-0096 – Possible Reactivation of Construction and Licensing Activities for the Watts Bar Nuclear Plant Unit 2.	ML072060688
IEEE Standard	
IEEE Std 603-2009, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations."	http://www.ieee.org or http://ibr.ansi.org/Standards/ieee.aspx

IEEE Std 603-2009 correction sheet, "Errata to IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations."	http://standards.ieee.org/findstds/errata/603-2009_errata.pdf

Throughout the development of this rulemaking, the NRC may post documents related to this rule, including public comments, on the Federal rulemaking Web site at http://www.regulations.gov under Docket ID NRC-2011-0089. The Federal rulemaking Web site allows you to receive alerts when changes or additions occur in a docket folder. To subscribe: 1) Navigate to the docket folder for NRC-2011-0089; 2) click the "Sign up for E-mail Alerts" link; and 3) enter your e-mail address and select how frequently you would like to receive e-mails (daily, weekly, or monthly).

IX. Voluntary Consensus Standards.

Section 12(d)(3) of the NTTAA, and implementing guidance in U.S. Office of Management and Budget (OMB) Circular A-119 (February 10, 1998), requires each Federal government agency (should it decide that regulation is necessary) to use a voluntary consensus standard instead of developing a government-unique standard. An exception to using a voluntary consensus standard is allowed where the use of such a standard is inconsistent with applicable law or is otherwise impractical. The NTTAA requires Federal agencies to use industry consensus standards to the extent practical; it does not require Federal agencies to endorse a standard in its entirety. Neither the NTTAA nor the OMB Circular A-119 prohibit an agency from adopting a voluntary consensus standard while taking exception to specific portions of the standard, if those portions are deemed to be "inconsistent with applicable law or otherwise impractical." Furthermore, taking specific exceptions furthers the Congressional intent of Federal reliance on voluntary consensus standards because it allows the adoption of substantial portions of consensus standards without the need to reject the standards in their entirety because of limited provisions that are not acceptable to the agency.

In this rulemaking, the NRC proposes to amend its regulations to incorporate by reference a more recent revision of IEEE Std 603, "Standard Criteria for Safety Systems for Nuclear Power Generating Stations." The IEEE Std 603-2009 is a national consensus standard developed by participants with broad and varied interests, in which all interested parties (including the NRC and licensees and designers of nuclear power plants) participate. In a Staff Requirements Memorandum (SRM) dated September 10, 1999, the Commission indicated its intent that a rulemaking identify all parts of an adopted voluntary consensus standard that are not adopted and justify not adopting such parts. The parts of IEEE Std 603-2009 that the NRC proposes to not adopt, partially adopt, or clarify to meet the NRC's regulations are identified in section III, "Discussion," and section XIV, "Backfitting and Issue Finality," in this document and in DG-1251. Information on DG-1251, including comments and supporting documentation, can be obtained by the methods identified in the ADDRESSES section of this document.

The justification for conditioning or not adopting parts of IEEE Std 603-2009 as set forth in these statements of consideration and the draft regulatory and backfit analysis for this proposed rule, satisfies the requirements of NTTAA, Section 12(d)(3), OMB Circular A–119, and the Commission's direction in the SRM dated September 10, 1999. In accordance with the NTTAA and OMB Circular A–119, the NRC is requesting public comment regarding whether other national or international consensus standards could be endorsed as an alternative to IEEE Std 603-2009.

X. Incorporation by Reference—Reasonable Availability to Interested Parties.

The NRC proposes to incorporate by reference IEEE Std 603-2009 and the correction sheet dated March 10, 2015, into the NRC's regulation in 10 CFR 50.55a. The author of IEEE Std 603-2009 is the Institute of Electrical and Electronics Engineers. As described in section III, “Discussion,” of this document, the IEEE Std 603-2009 addresses the functionality and design requirements for power, instrumentation, and control systems for nuclear power plants.

The NRC is required by law to obtain approval for incorporation by reference from the Office of the Federal Register (OFR). The OFR’s requirements for incorporation by reference are set forth in 1 CFR part 51. On November 7, 2014, the OFR adopted changes to its regulations governing incorporation by reference (79 FR 66267). The OFR regulations require an agency to include in a proposed rule a discussion of the ways that the materials the agency proposes to incorporate by reference are reasonably available to interested parties or how it worked to make those materials reasonably available to interested parties. The discussion in this section complies with the requirement for proposed rules as set forth in 1 CFR 51.5(a)(1).

The NRC considers “interested parties” to include all potential NRC stakeholders, not only the individuals and entities regulated or otherwise subject to the NRC’s regulatory oversight. These NRC stakeholders are not a homogenous group but vary with respect to the considerations for determining reasonable availability. Therefore, the NRC distinguishes between different classes of interested parties for purposes of determining whether the material is “reasonably available.” The NRC considers the following to be classes of interested parties in NRC rulemakings with regard to the material to be incorporated by reference:

- Individuals and small entities regulated or otherwise subject to the NRC’s regulatory oversight (this class also includes applicants and potential applicants for licenses and other NRC regulatory approvals) and who are subject to the material to be incorporated by reference by rulemaking. In this context, “small entities” has the same meaning as a “small entity” under 10 CFR 2.810.
- Large entities otherwise subject to the NRC’s regulatory oversight (this class also includes applicants and potential applicants for licenses and other NRC regulatory approvals) and who are subject to the material to be incorporated by reference by rulemaking. In this

context, “large entities” are those which do not qualify as a “small entity” under 10 CFR 2.810.

- Non-governmental organizations with institutional interests in the matters regulated by the NRC.

- Other Federal agencies, states, local governmental bodies (within the meaning of 10 CFR 2.315(c)).

- Federally-recognized and State-recognized[2] Indian tribes.

- Members of the general public (i.e., individual, unaffiliated members of the public who are not regulated or otherwise subject to the NRC’s regulatory oversight) who may wish to gain access to the materials which the NRC proposes to incorporate by reference by rulemaking in order to participate in the rulemaking.

The NRC makes the materials to be incorporated by reference available for inspection to all interested parties, by appointment, at the NRC Technical Library, which is located at Two White Flint North, 11545 Rockville Pike, Rockville, Maryland 20852; telephone: 301-415-7000; e-mail: Library.Resource@nrc.gov.

The IEEE makes IEEE Std 603-2009 available online to the public in read-only format without cost at http://ibr.ansi.org/Standards/ieee.aspx, which is a Web site maintained by the American National Standards Institute (ANSI). The IEEE Std 603-2009 correction sheet dated March 10, 2015, is available online without cost at http://standards.ieee.org/findstds/errata/603-2009_errata.pdf. Therefore, all classes of potentially interested parties (as previously stated in this section) are able to read the text of IEEE Std 603-2009 online via the Internet.

Because access to IEEE Std 603-2009 and the correction sheet dated March 10, 2015, are available in various forms for no cost from several sources and in several different ways (e.g., through read-only online access and public inspection), the NRC determines that IEEE

[2] State-recognized Indian tribes are not within the scope of 10 CFR 2.315(c). However, for purposes of the NRC’s compliance with 1 CFR 51.5, “interested parties” includes a broad set of stakeholders, including State-recognized Indian tribes.

Std 603-2009 and the correction sheet dated March 10, 2015, are reasonably available to all interested parties.

XI. Finding of No Significant Environmental Impact: Environmental Assessment.

The Commission has determined under the National Environmental Policy Act of 1969, as amended, and the NRC's regulations in subpart A of 10 CFR part 51, that this rule, if adopted, would not be a major Federal action significantly affecting the quality of the human environment and, therefore, an environmental impact statement is not required. The basis for this determination reads as follows:

Identification of the Proposed Action

This proposed action is in accordance with the NRC's policy to incorporate by reference in § 50.55a(h) a revised IEEE standard to provide updated rules for design, modifying, or replacing protection systems and safety systems in nuclear power plants. The proposed rule mandates the use of IEEE Std 603-2009 and the correction sheet dated March 10, 2015, for future nuclear power plants, including final design approvals, design certifications, combined licenses, and manufacturing licenses approved by the NRC under 10 CFR part 52. Further, licensees of currently operating nuclear power plants may continue to meet the requirements stated in the edition or revision of the standard in effect on the formal date of their application for a construction permit or may, at their option, use IEEE Std 603-2009 and the correction sheet dated March 10, 2015, provided licensees follow all applicable requirements for making changes to their nuclear power plant licensing basis. However, applications for modifying or adding to existing protection systems and safety systems or installing protection systems, protection system functions, safety systems, or safety system functions that add or remove safety functionality, change the technology of the protection system or safety system, or change the diversity strategy in the protection system or safety system on or after the effective date of this rule would be required to meet the requirements stated in IEEE Std 603-2009 and the correction sheet dated March 10, 2015.

The Need for the Proposed Action

This action is needed to ensure that an adequate level of safety is maintained in current and future nuclear power plants whenever modifications or additions to existing protection systems and safety systems, or installations of protection systems, protection system functions, safety systems, or safety system functions are made in which new safety functionality, a technology change, or a change in diversity strategy is implemented. This action also adds appropriate cross references in the updated regulation.

Environmental Impacts of the Proposed Action

The proposed rulemaking will not significantly increase the probability or consequences of accidents. No changes are being made that could affect land use, water use, air resources, aquatic or terrestrial ecology, threatened, endangered and protected species, essential fish habitats, or historical or cultural resources. No changes are being made in the types of effluents that may be released off-site; and there is no significant increase in public radiation exposure. The NRC estimates the radiological dose to plant personnel implementing the requirements of this new rule will be no more than that experienced under the current regulations. Therefore, the NRC concludes that any increase in occupational exposure would not be significant. The proposed rulemaking does not involve non-radiological plant effluents and has no other environmental impact. Therefore, no significant non-radiological impacts are associated with the proposed rule.

Environmental Impacts of Alternatives to the Proposed Action

The primary alternative to the proposed action would be to maintain the existing incorporation by reference of the existing IEEE Std 603-1991. This alternative would result in no change in current environmental impacts. The environmental impacts of the proposed action and the alternative are similar.

Alternative Use of Resources and Consultation

Since no difference in environmental impacts result from the proposed action and the alternative, there is no need to evaluate alternative use of resources. Likewise, there is no need to perform consultative activities. The NRC has sent a notification of this proposed rule to every State Liaison Officer and requested their comments on the environmental assessment contained herein.

The determination of this environmental assessment is that there will be no significant offsite impact to the public from this action. However, the general public should note that the NRC is seeking public participation. Comments on any aspect of the environmental assessment may be submitted to the NRC as indicated in the ADDRESSES section of this document.

XII. Paperwork Reduction Act Statement.

This proposed rule contains new or amended collections of information subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq). This proposed rule has been submitted to the Office of Management and Budget for review and approval of the information collections.

Type of submission, new or revision: Revision.

The title of the information collection: Incorporation by Reference of Institute of Electrical and Electronics Engineers Standard 603-2009.

The form number if applicable: Not applicable.

How often the collection is required or requested: On occasion.

Who will be required or asked to respond: Nuclear reactor licensees and applicants.

An estimate of the number of annual responses: -0.5 responses (reduction of responses annually).

The estimated number of annual respondents: 1 respondent every 6 years.

An estimate of the total number of hours needed annually to comply with the information collection requirement or request: -50 hours (reduction of reporting hours).

Abstract: The NRC proposes to amend its regulations to incorporate by reference the IEEE Std 603-2009, "Standard Criteria for Safety Systems for Nuclear Power Generating Stations," which establishes minimal functional and design requirements for power, instrumentation, and control systems for nuclear power plants. The proposed rule would affect applicants for new reactor designs and licensees of currently operating nuclear power plants who apply for a license or a license amendment after the effective date of this rule. The proposed rule would reduce licensee burden because licensees would no longer need to submit alternative requests in order to use this updated, more current standard.

The U.S. Nuclear Regulatory Commission is seeking public comment on the potential impact of the information collections contained in this proposed rule and on the following issues:

1. Is the proposed information collection necessary for the proper performance of the functions of the NRC, including whether the information will have practical utility?

2. Is the estimate of the burden of the proposed information collection accurate?

3. Is there a way to enhance the quality, utility, and clarity of the information to be collected?

4. How can the burden of the proposed information collection on respondents be minimized, including the use of automated collection techniques or other forms of information technology?

A copy of the OMB clearance package and proposed rule is available in ADAMS (Accession Nos. ML14114A532 and ML113190983) or may be viewed free of charge at the NRC's PDR, One White Flint North, 11555 Rockville Pike, Room O-1 F21, Rockville, MD 20852. You may obtain information and comment submissions related to the OMB clearance package by searching on http://www.regulations.gov under Docket ID NRC-2011-0089.

You may submit comments on any aspect of these proposed information collection(s), including suggestions for reducing the burden and on the above issues, by the following methods:

- **Federal rulemaking Web Site:** Go to http://www.regulations.gov and search for Docket ID NRC-2011-0089.

- **Mail comments to:** FOIA, Privacy, and Information Collections Branch, Office of Information Services, Mail Stop: T-5 F53, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001 or to Vlad Dorjets, Desk Officer, Office of Information and Regulatory Affairs (3150-0011), NEOB-10202, Office of Management and Budget, Washington, DC 20503; telephone 202-395-7315, e-mail: oira_submission@omb.eop.gov.

Submit comments by **[INSERT DATE 30 DAYS AFTER PUBLICATION IN THE *FEDERAL REGISTER*]**. Comments received after this date will be considered if it is practical to do so, but the NRC staff is able to ensure consideration only for comments received on or before this date.

Public Protection Notification

The NRC may not conduct or sponsor, and a person is not required to respond to, a request for information or an information collection requirement unless the requesting document displays a currently valid OMB control number.

XIII. Regulatory Analysis: Availability.

The NRC has prepared a draft regulatory analysis on this proposed rule (ADAMS Accession No. ML120310194). The analysis examines the costs and benefits of the alternatives considered by the NRC. The analysis concluded that the proposed rule relative to the regulatory baseline is cost-benefit neutral for industry with an estimate net cost of $7,000 based on a 7-percent net present value to a net benefit of $26,000 based on a 3-percent net present value. For the NRC, the proposed rule is not quantitatively cost beneficial, although, as discussed below, there are significant benefits that were not quantified in this analysis. The quantified costs for the NRC range from an estimated net cost of $372,000 based on a 7% net present value to a net cost of $355,000 based on a 3% net present value. The NRC benefits from the proposed rulemaking because of the averted cost savings resulting from the reduction in the number of alternative requests on a plant-specific basis under 10 CFR 50.55a(z). The NRC requests public comments on the draft regulatory analysis. Comments on the draft regulatory analysis may be submitted to the NRC by any method provided in the ADDRESSES section of this document.

XIV. Backfitting and Issue Finality.

Introduction

The proposed rule's substantive provisions, in § 50.55a(h), would apply to the design of protection and safety systems for currently-operating nuclear power reactors, as well as designs for future nuclear power reactors, and would affect different classes of NRC licenses and regulatory approvals. Backfitting and issue finality for each of the affected classes of licenses and regulatory approvals is discussed in the following paragraphs.

Construction Permits

Currently, there are three construction permits in effect: the Tennessee Valley Authority (TVA) Watts Bar Nuclear Plant, Unit 2, which is active, and the TVA Bellefonte Nuclear Plant, Units 1 and 2, which are in deferral status. The proposed rule would apply to the Watts Bar Nuclear Plant, Unit 2, and the Bellefonte Nuclear Plant, Units 1 and 2, but only if the construction permit holder makes changes or modifications to, or replaces the plant's protection system or safety system (as reviewed and approved in the construction permit application and described in the preliminary safety analysis reports) under § 50.55a(h)(3) of the proposed rule. Inasmuch as such proposed changes, modifications, or replacements would be a voluntary action initiated by the construction permit holder, the imposition of the proposed rule's requirements in that circumstance does not constitute backfitting as defined in § 50.109(a)(1). As discussed earlier in § 50.55a(h)(2)(ii), the NRC is not requiring either Watts Bar Nuclear Plant, Unit 2, or Bellefonte Nuclear Plant, Units 1 and 2, to meet current requirements applicable to newly licensed nuclear power plants.

The proposed rule would apply to all newly-applied for construction permits. Imposition of the proposed rule does not constitute backfitting, inasmuch as the backfit rule does not protect either a current applicant or a future (prospective) applicant.

Operating Licenses

The proposed rule would apply to the 99 operating nuclear power reactors licensed under 10 CFR part 50, but only insofar as the plant's currently-approved protection system or safety system may be modified or replaced in the future and therefore is subject to § 50.55a(h)(3) of the proposed rule. Inasmuch as such proposed changes, modifications, or replacements would be a voluntary action initiated by the licensee, the imposition of the proposed rule's requirements in that circumstance does not constitute backfitting as defined in § 50.109(a)(1).

Currently, there is only one application for an operating license in process before the NRC; this application is for TVA's Watts Bar Nuclear Plant, Unit 2. The proposed rule would apply to Watts Bar Nuclear Plant, Unit 2, operating license, except for matters that were previously approved in the Watts Bar Nuclear Plant, Unit 2, construction permit. Thus, the "mandatory compliance" provisions of the proposed rule, § 50.55a(h)(3), would apply to the Watts Bar Nuclear Plant, Unit 2, operating license. Imposition of the proposed rule on Watts Bar Nuclear Plant, Unit 2, would not constitute backfitting, inasmuch as the backfit rule does not protect a current applicant. In addition, the "mandatory compliance" provisions of the proposed rule would not constitute backfitting inasmuch as those provisions apply to voluntary actions to change the plant's licensing basis that may be initiated by the licensee.

The proposed rule would apply to all new applications for operating licenses. Imposition of the proposed rule on future applications for operating licenses does not constitute backfitting, inasmuch as the backfit rule does not protect a future (prospective) applicant. In addition, the “mandatory compliance” provisions in § 50.55a(h)(3) of the proposed rule would not constitute backfitting inasmuch as those provisions apply to voluntary actions to change the plant’s licensing basis that may be initiated by the licensee.

Combined Licenses

The proposed rule would apply to a combined license that does not reference a standard design certification or manufacturing license. Currently, there are no manufacturing licenses issued under 10 CFR part 52, and no combined licenses issued that do not reference a standard design certification (the combined licenses issued by the NRC for the Vogtle Electric Generating Plant, Units 3 and 4, and the combined licenses issued for the Virgil C. Summer, Units 2 and 3, reference the AP1000 standard design certification rule, 10 CFR part 52, appendix D, as amended (76 FR 82079; December 30, 2011). The combined license issued to the Enrico Fermi Nuclear Plant Unit 3 references the Economic Simplified Boiling Water Reactor standard design. With respect to future combined license or manufacturing license applicants that do not reference a standard design certification or manufacturing license, the Backfit Rule and the issue finality provisions in 10 CFR part 52 do not protect a future (prospective) applicant.

The proposed rule would apply to current (as of the date of the final IEEE rulemaking) and future combined licenses referencing a standard design certification or manufacturing license, but only if the combined license applicant or holder either: 1) seeks an exemption or departure from the referenced design certification rule’s safety system, or 2) modifies or replaces the safety system and therefore is subject to § 50.55a(h)(3) of the proposed rule. The NRC notes that the NRC’s approval of a certified design includes all aspects of the reactor’s design that must be designed to the relevant IEEE standard under § 50.55a(h), and the combined license applicant and holder has no further responsibility to address the adequacy of the electrical design for the safety system. Hence the proposed rule does not directly apply to such combined license applicants and holders. As of this rulemaking, there are combined licenses for the Vogtle Electric Generating Plant, Units 3 and 4, and the combined licenses issued for the Virgil C. Summer, Units 2 and 3, both of which reference the AP1000 standard design certification rule as well as a combined license for Enrico Fermi Nuclear Plant Unit 3 which references the Economic Simplified Boiling Water Reactor standard design.

Imposition of the proposed rule in the first circumstance (seeking a departure or an exemption from a referenced design certification rule) does not constitute backfitting because seeking such a departure or exemption would be a voluntary action initiated by the applicant or licensee, and imposition of the proposed rule's requirements in this circumstance does not constitute backfitting as defined in § 50.109(a)(1), nor is the proposed rule inconsistent with any of the issue finality provisions in §§ 52.63, 52.83, 52.98 or the currently-approved design certifications in 10 CFR part 52, appendices A through E.

The second circumstance (modifying or replacing a safety system) is also a voluntary action initiated by the applicant or licensee, and imposition of the proposed rule's requirements in this circumstance does not constitute backfitting as defined in § 50.109(a)(1), nor is the proposed rule inconsistent with any of the issue finality provisions in §§ 52.63, 52.83, 52.98 or the currently-approved design certifications in 10 CFR part 52, appendices A through E.

The proposed rule would also apply to any portion of a safety system (within the meaning of § 50.55a and IEEE Std 603-2009) of currently-issued combined licenses referencing design certifications that are outside the scope of the referenced design certification (including exemption and departure requests). For those portions of safety systems outside the scope of the referenced standard design certification, the combined license would be subject to the "mandatory compliance" provisions in § 50.55a(h)(3) of the proposed rule. This does not constitute backfitting, inasmuch as the proposed rule would not mandate changes to the currently-approved design of any safety systems outside the scope of the referenced design certification to comply with IEEE Std 603-2009 and the correction sheet dated March 10, 2015. Rather, only future, licensee-initiated changes to any safety systems outside the scope of the referenced design would be required to meet the requirements in IEEE Std 603-2009 and the correction sheet dated March 10, 2015, under any of the circumstances set forth in § 50.55(h)(3). The NRC does not consider voluntary, licensee-initiated changes to the licensing basis to be “imposed,” and such changes, therefore, do not constitute backfitting under § 50.109(a)(1).

The proposed rule would apply to future combined license applicants that reference a standard design certification or manufacturing license, in the same manner as current holders of combined licenses referencing a standard design certification, as explained in the previous paragraphs. This § 50.55a rulemaking mandating the use of IEEE Std 603-2009 and the correction sheet dated March 10, 2015, for future combined licenses, referencing standard design certifications, issued after the effective date of this rule does not constitute backfitting, because these requirements are prospective in nature and effect. The backfit rule and the issue finality provisions in 10 CFR part 52 do not protect a future (prospective) applicant. The backfit rule and the issue finality provisions of 10 CFR part 52 were not intended to apply to every NRC action that substantially changes the expectations of future applicants under 10 CFR part 52.

Standard Design Certifications

The proposed rule would apply to the currently-approved standard design certifications in 10 CFR part 52, appendices A through E (and any future standard design certification that may be approved before the issuance of the final § 50.55a rulemaking incorporating by reference IEEE Std 603-2009), but only if the design of the safety system for the certification is modified or changed in a subsequent amendment to the design certification rule. Regardless of whether the amendment is sought by an applicant or is initiated by the NRC, the issue finality provisions of § 52.63 would have to be satisfied as part of that amendment rulemaking.

The proposed rule would apply to all standard design certification applications active at the time of the final § 50.55a rulemaking incorporating by reference IEEE Std 603-2009 and the correction sheet dated March 10, 2015, as well as all future applications for standard design certifications. Imposition of the proposed rule on current or future standard design certification applicants does not constitute backfitting as defined in § 50.109 nor is it inconsistent with § 52.63 (the issue finality provisions applicable to design certifications in 10 CFR part 52), because neither the backfit rule nor § 52.63 protect a current or future (prospective) design certification applicant.

Manufacturing Licenses

There are no current applicants for, or holders of, manufacturing licenses under 10 CFR part 52, subpart F. The proposed rule would apply to future applications for manufacturing licenses. Imposing the proposed rule on future applicants for manufacturing licenses does not constitute backfitting as defined in § 50.109 nor is it inconsistent with § 52.171 (issue finality provisions applicable to manufacturing licenses in 10 CFR part 52) because neither the backfit rule nor § 52.171 protects a future (prospective) manufacturing license applicant.

Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors

The proposed rule would add a reference to sections 5.3 and 5.4 of IEEE Std 603-2009 in § 50.69(b)(1)(v). Inasmuch as compliance with § 50.69(b)(1)(v) would be a voluntary action initiated by the licensee or applicant, the imposition of the proposed rule's requirements in that circumstance does not constitute backfitting as defined in § 50.109(a)(1).

Emergency response data systems

The proposed rule would add additional isolation requirements for emergency response data systems in 10 CFR part 50, appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities."

The proposed rule would not require licensees and applicants to address communication independence in addition to electrical independence for emergency response data systems for currently operating nuclear plants because communications from the ERDS to safety systems does not exist in these plants. Therefore, no action is required of licensees to implement communication independence. Further, the proposed rule would not require holders of combined licenses, standard design certifications, and manufacturing licenses for the reasons stated in the above respective sections. Therefore, imposing the proposed rule on future applicants for combined licenses, standard design certifications, and manufacturing licenses does not constitute backfitting as defined in § 50.109 and applicable sections of 10 CFR part 52.

XV. Regulatory Flexibility Certification.

In accordance with the Regulatory Flexibility Act (5 U.S.C. 605(b)), the Commission certifies that this rule will not, if promulgated, have a significant economic impact on a substantial number of small entities. This proposed rule affects only the licensing and operation of nuclear power plants. The companies that own these plants do not fall within the scope of the definition of "small entities" set forth in the Regulatory Flexibility Act or the size standards established by the NRC (10 CFR 2.810).

List of Subjects in 10 CFR Part 50

Antitrust, Classified information, Criminal penalties, Fire protection, Incorporation by reference, Intergovernmental relations, Nuclear power plants and reactors, Radiation protection, Reactor siting criteria, Reporting and recordkeeping requirements.

For the reasons set out in the preamble and under the authority of the Atomic Energy Act of 1954, as amended; the Energy Reorganization Act of 1974, as amended; and 5 U.S.C. 552 and 553, the NRC is proposing to adopt the following amendments to 10 CFR part 50.

PART 50 -- DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

1. The authority citation for part 50 continues to read as follows:

AUTHORITY: Atomic Energy Act secs. 102, 103, 104, 105, 147, 149, 161, 181, 182, 183, 186, 189, 223, 234 (42 U.S.C. 2132, 2133, 2134, 2135, 2167, 2169, 2201, 2231, 2232, 2233, 2236, 2239, 2273, 2282); Energy Reorganization Act secs. 201, 202, 206 (42 U.S.C. 5841, 5842, 5846); Nuclear Waste Policy Act sec. 306 (42 U.S.C. 10226); Government Paperwork Elimination Act sec. 1704 (44 U.S.C. 3504 note); Energy Policy Act of 2005, Pub. L. No. 109-58, 119 Stat. 194 (2005). Section 50.7 also issued under Pub. L. 95-601, sec. 10, as amended by Pub. L. 102-486, sec. 2902 (42 U.S.C. 5851). Section 50.10 also issued under Atomic Energy Act secs. 101, 185 (42 U.S.C. 2131, 2235); National Environmental Policy Act sec. 102 (42 U.S.C. 4332). Sections 50.13, 50.54(d), and 50.103 also issued under Atomic Energy Act sec. 108 (42 U.S.C. 2138).

Sections 50.23, 50.35, 50.55, and 50.56 also issued under Atomic Energy Act sec. 185 (42 U.S.C. 2235). Appendix Q also issued under National Environmental Policy Act sec. 102 (42 U.S.C. 4332). Sections 50.34 and 50.54 also issued under sec. 204 (42 U.S.C. 5844). Sections 50.58, 50.91, and 50.92 also issued under Pub. L. 97-415 (42 U.S.C. 2239). Section 50.78 also issued under Atomic Energy Act sec. 122 (42 U.S.C. 2152). Sections 50.80 - 50.81 also issued under Atomic Energy Act sec. 184 (42 U.S.C. 2234).

2. In § 50.55a, revise paragraphs (a)(2)(iii) and (a)(2)(iv), add new paragraphs (a)(2)(v) and (a)(2)(vi), and revise paragraph (h) to read as follows:

§ 50.55a Codes and standards.

(a) * * *

(2) * * *

(iii) *IEEE standard 603-1991.* (IEEE Std 603-1991), “Standard Criteria for Safety Systems for Nuclear Power Generating Stations” (Approval Date: June 27, 1991), referenced in paragraph (h)(2) of this section. All other standards that are referenced in IEEE Std 603-1991 are not approved for incorporation by reference.

(iv) *IEEE standard 603-1991, correction sheet.* (IEEE Std 603-1991 correction sheet), “Standard Criteria for Safety Systems for Nuclear Power Generating Stations, Correction Sheet, Issued January 30, 1995, ” referenced in paragraph (h)(2) of this section. (This correction sheet is available from IEEE at http://standards.ieee.org/findstds/errata/.)

(v) *IEEE standard 603-2009.* (IEEE Std 603-2009), “IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations” (Approval Date: November 5, 2009), referenced in paragraphs (h)(2) and (3) of this section. All other standards that are referenced in IEEE Std 603-2009 are not approved for incorporation by reference.

(vi) *IEEE standard 603-2009, correction sheet.* (IEEE Std 603-2009 correction sheet), "Errata to IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations" (Issued: March 10, 2015), referenced in paragraphs (h)(2) and (3) of this section.

* * * * *

(h) *Protection and safety systems*. Protection systems and safety systems of nuclear power plants must meet the requirements in this paragraph.

(1) *Definitions*. As used in paragraph (h) of this section,

Current reactors means nuclear power plants whose construction permits were issued before May 13, 1999.

New reactors means design certifications; standard design approvals; manufacturing licenses; and combined licenses not referencing a design certification, standard design approval, or manufacturing license under 10 CFR part 52 issued on or after the effective date of the final rule; construction permits and operating licenses under 10 CFR part 50 issued on or after the effective date of the final rule, except for an applicant for an operating license who received a construction permit for that facility before the effective date of the final rule; and holders of combined licenses issued under 10 CFR part 52 before the effective date of the final rule, but only if the combined license holder voluntarily modifies its data communication independence strategy.

(2)(i) *Nuclear power plant construction permits issued before January 1, 1971*. The protection system of a nuclear power plant whose construction permit was issued before January 1, 1971, must be either consistent with the plant's licensing basis; or meet the requirements in IEEE Std 603-1991, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations," and the correction sheet dated January 30, 1995, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations Correction Sheet Issued January 30, 1995."

(ii) *Nuclear power plant construction permits issued after January 1, 1971, but before May 13, 1999.* The protection system of a nuclear power plant whose construction permit was issued after January 1, 1971, but before May 13, 1999, must meet the requirements in IEEE Std 279-1968, "Proposed IEEE Criteria for Nuclear Power Plant Protection Systems," or the requirements in IEEE Std 279-1971, "IEEE Standard: Criteria for Protection Systems for Nuclear Power Generating Stations," or the requirements in IEEE Std 603-1991, "Standard Criteria for Safety Systems for Nuclear Power Generating Stations," and the correction sheet dated January 30, 1995, "Standard Criteria for Safety Systems for Nuclear Power Generating Stations Correction Sheet Issued January 30, 1995."

(iii) *Standard design certifications issued before May 13, 1999.* The protection system of a standard design certification issued before May 13, 1999, must meet the requirements in IEEE Std 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations."

(iv) *Standard design certifications issued after May 13, 1999, but before* ***[EFFECTIVE DATE OF THIS RULE]***. Safety systems in standard design certifications issued after May 13, 1999, but before **[EFFECTIVE DATE OF THIS RULE]**, must meet the requirements in IEEE Std 603-1991, "Standard Criteria for Safety Systems for Nuclear Power Generating Stations," and the correction sheet dated January 30, 1995, "Standard Criteria for Safety Systems for Nuclear Power Generating Stations Correction Sheet Issued January 30, 1995." If a combined license or manufacturing license references a standard design certification, then the safety system for the licensed facility must comply with those applicable regulations stated in the referenced standard design certification.

(v) *Standard design certifications issued after* ***[EFFECTIVE DATE OF THIS RULE]****.* Safety systems in standard design certifications under 10 CFR part 52 issued after **[EFFECTIVE DATE OF THIS RULE]** must meet the requirements in IEEE Std 603-2009 and the correction sheet dated March 10, 2015, subject to the conditions in paragraph (h)(4) through paragraph (8).

(vi) *Applications for nuclear power plant construction permits submitted after* ***[EFFECTIVE DATE OF THIS RULE]*** *under 10 CFR part 50.* Safety systems in construction permits under 10 CFR part 50 for applications submitted after **[EFFECTIVE DATE OF THIS RULE]** must meet the requirements in IEEE Std 603-2009 and the correction sheet dated March 10, 2015, subject to the conditions in (h)(4) through paragraph (8).

(vii) *Nuclear power plant combined licenses and manufacturing licenses under 10 CFR part 52 issued after* ***[EFFECTIVE DATE OF THIS RULE]****.* Safety systems in combined licenses and manufacturing licenses issued after **[EFFECTIVE DATE OF THIS RULE]** must meet the requirements in IEEE Std 603-2009 and the correction sheet dated March 10, 2015, subject to the conditions in (h)(4) through paragraph (8) of this section, *provided, however*, that if the combined licenses or manufacturing license reference an approved standard design certification, then the safety system must comply with those applicable regulations stated in the referenced standard design certification.

(3) *Modifications and replacements of protection systems and safety systems*. Modifications to and replacements of protection systems and safety systems must meet the requirements stated in this section. If a modification or replacement changes the functionality, technology (including changes to equipment qualification characteristics), independence strategy, or diversity strategy in a protection system or safety system, then the changed or replaced components, functions, or systems must meet the requirements in IEEE Std 603-2009 and the correction sheet dated March 10, 2015, subject to the conditions in paragraph (h)(4) through paragraph (8) of this section. If this modification or replacement does not cause these changes in a protection system or safety system, then the changed or replaced components, functions, or systems may meet the requirements in the existing licensing basis.

(4) *System Integrity.* When addressing the requirements in section 5.5 of IEEE Std 603-2009, safety system functions must be demonstrated to be both repeatable and predictable.

(5) *Independence.* The following requirements must be met when addressing the requirements in section 5.6 of IEEE Std 603-2009:

(i) *Independence between redundant portions of a safety system.* The safety system architecture must incorporate independence between redundant portions of a safety system. Independence in the safety system architecture must be analyzed to address: safety system internal and external hazards, the extent of interconnectivity between redundant portions of the safety system, and the impact of failures or degradation in one portion of a safety system on the ability of redundant safety system portions to accomplish the safety functions.

(ii) *Independence between safety systems and other systems.* When applying IEEE Std 603-2009 section 5.6.3.1.a.2.ii and section 5.6.3.1.b, independence must exist between safety systems and other systems for all signal technologies. Independence between safety systems and other systems shall be analyzed to address: hazards posed by other systems on the safety system, the extent of interconnectivity between the safety system and other systems, and the impact of failures or degradation in other systems on the ability of the safety system to accomplish the safety functions.

(iii) *Detailed criteria.* The following conditions apply to section 5.6 of IEEE Std 603-2009.

(A) Signals between redundant safety divisions and signals from a non-safety-related system to a safety division must be processed in a manner that does not impair the safety functions of any safety system division.

(B) Safety system divisions must detect and mitigate signal faults and failures received from outside the safety system division in a manner that does not impair the safety system safety functions of the division.

(C) For current reactors, communications or signals from outside the safety division during operation must support safety or provide a safety benefit.

(D) For new reactors:

(1) Data communications between safety and non-safety systems must be one-way, accomplished by a physical mechanism, from safety to non-safety systems while the affected portion of the safety system is in operation.

(2) Signals may be shared between redundant portions of safety systems only if the signals are required to perform a safety function.

(3) A safety system may receive signals from non-safety systems while the safety system is in operation only if the received signal supports diversity or automatic anticipatory reactor trip functions. These signals must be transmitted over a hardwired connection using means other than data communication.

(4) Applicants for design certifications, standard design approvals, or manufacturing licenses who propose an alternative under paragraph (z) of this section for complying with the requirement in paragraph (h)(5) of this section with respect to data communications independence shall identify both direct and indirect communication pathways to safety systems from other systems.

(6) *Retaining safety function capability during maintenance bypass.* The constraints referenced in IEEE Std 603-2009 section 6.5.1.b are the constraints described in section 6.7, "Maintenance Bypass."

(7) *Maintenance bypass*. The maintenance bypass requirements in section 6.7 of IEEE Std 603-1991 must be met instead of the requirements in section 6.7 of IEEE Std 603-2009.

(8) *Documentation supporting compliance.* Applicants and licensees shall develop and maintain documentation, analyses, and design details demonstrating compliance with paragraphs (h)(2) through (7) of this section.

* * * * *

3. In § 50.69, revise paragraph (b)(1)(v) to read as follows:

§ 50.69 Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors.

* * * * *

(b) * * *

(1) * * *

(v) The inservice testing requirements in § 50.55a(f); the inservice inspection, and repair and replacement (with the exception of fracture toughness), requirements for the American Society of Mechanical Engineers Class 2 and Class 3 SSCs in § 50.55a(g); and the electrical component quality and qualification requirements in sections 4.3 and 4.4 of IEEE Std 279-1971, sections 5.3 and 5.4 of IEEE Std 603-1991, and sections 5.3 and 5.4 of IEEE Std 603-2009, as incorporated by reference in § 50.55a(a).

* * * * *

Appendix E to Part 50 -- [Amended]

4. In appendix E to part 50, revise footnote 7 to remove the words "Protection Systems" and add, in its place, the words "Protection and safety systems."

Dated at Rockville, Maryland, this day of , 2015.

For the Nuclear Regulatory Commission.

Annette L. Vietti-Cook,
Secretary of the Commission.

Appendix E to Part 50 -- [Amended]

4. In appendix E to part 50, revise footnote 7 to remove the words “Protection Systems” and add, in its place, the words “Protection and safety systems.”

Dated at Rockville, Maryland, this day of , 2015.

For the Nuclear Regulatory Commission.

Annette L. Vietti-Cook,
Secretary of the Commission.

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DATE	12/31/2014	11/5/2014	10/27/2014	10/27/2014	10/28/2014
OFFICE	NRR/DE*	NRR/DE*	NRR/DE*	NRR/DE*	NRR/DE*
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DATE	11/6/2014	11/10/2014	10/27/2014	11/14/2014	10/23/2014
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