U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

APPENDIX 7.1-B GUIDANCE FOR EVALUATION OF CONFORMANCE TO IEEE Std 279

REVIEW RESPONSIBILITIES

Primary - Organization responsible for the review of instrumentation and controls

Secondary - None

Review Note: The revision numbers of Regulatory Guides (RG) and the years of endorsed industry standards referenced in this Standard Review Plan (SRP) section are centrally maintained in SRP Section 7.1-T, "Regulatory Requirements, Acceptance Criteria, and Guidelines for Instrumentation and Control Systems Important to Safety," (Table 7-1). Therefore, the individual revision numbers of RGs (except RG 1.97) and years of endorsed industry standards are not shown in this section. References to industry standards incorporated by reference into regulation (IEEE Std 279-1971 and IEEE Std 603-1991) and industry standards that are not endorsed by the agency do include the associated year in this section. See Table 7-1 to ensure that the appropriate RGs and endorsed industry standards are used for the review.

Revision 6 – August 2016

1. AREAS OF REVIEW

For nuclear power plants with construction permits issued before January 1, 1971, Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(h), "Protection and Safety Systems," requires that protection systems must be consistent with their licensing basis or may meet the requirements of the Institute of Electrical and Electronics Engineers (IEEE) Standard (Std) 603-1991, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations," and the correction sheet dated January 30, 1995. For nuclear power plants with construction permits issued after January 1, 1971, but before May 13, 1999, 10 CFR 50.55a(h) requires that protection systems meet the requirements of the IEEE Std 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations," or IEEE Std 603-1991 and the correction sheet dated January 30, 1995. The scope of IEEE Std 279-1971 includes those systems that actuate a reactor trip, and that in the event of a serious reactor accident, actuate engineered safety features. This appendix discusses the requirements of IEEE Std 279-1971, Clauses 3 and 4, as they are used in the review of the reactor trip systems (RTS) and engineered safety features actuation systems (ESFAS) to determine that these systems meet the U.S. Nuclear Regulatory Commission (NRC) regulations. Although required by NRC regulations only for protection systems, the criteria of IEEE Std 279-1971 address considerations such as design bases, redundancy, independence, single failures, qualification, bypasses, status indication, and testing that may be used as review guidance, where appropriate, for any instrumentation and control (I&C) system, as elaborated in SRP Sections 7.2 through 7.9. Therefore, for I&C systems not a part of the protection system, but having a high degree of importance to safety, the reviewer may use the concepts of IEEE Std 279-1971 for the review of these systems. SRP, Appendix 7.1-C provides guidance for evaluating conformance to IEEE Std 603-1991.

SRP Appendix 7.1-D, "Guidance for Evaluation of the Application of IEEE Std 7-4.3.2," provides guidance for evaluating conformance to the acceptance criteria contained in RG 1.152, "Criteria for Digital Computers in Safety Systems of Nuclear Power Plants," which endorses IEEE Std 7-4.3.2, "IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations."

2. SCOPE

This appendix discusses the requirements of IEEE Std 279-1971 as they are used in the review of protection systems; however, it is not intended to be a stand-alone document. Each subsection of this appendix relates directly to one or more clauses of the standard. Additional background or detailed information relevant to this review can be found in the references to this appendix.

A review of protection systems by the organization responsible for the review of I&Cs that follows the guidance of IEEE Std 279-1971 should be coordinated with other organizations as appropriate to address the following considerations:

- Many of the auxiliary supporting features and other auxiliary features are described in Chapters 4, 5, 6, 8, 9, 10, 12, 15, 18 and 19 of the safety analysis report (SAR). The reviewers from the organization responsible for the review of I&C should coordinate with the reviewers of these SAR sections to ensure that auxiliary features are appropriately addressed by the review.

- The site characteristics, systems (both physical and administrative), and analyses described in the other sections of the SAR may impose requirements on the I&C systems. The reviewers from the organization responsible for the review of I&C should coordinate with the reviewers of these sections of the SAR to ensure the I&C systems appropriately address these requirements.

- I&C systems may impose requirements upon other plant systems and analyses. The reviewers from the organization responsible for the review of I&C should coordinate with the reviewers of the affected systems to ensure that the reviewers are aware of these requirements.

- Other plant systems will impose requirements on the I&C systems. The reviewers from the organization responsible for the review of I&Cs should coordinate with the reviewers of the interfacing systems to ensure that these requirements are considered in the review. The coordination review needed for each I&C system is discussed in SRP Section 7.0.

3. DESIGN BASIS

Clause 3 of IEEE Std 279-1971 requires in part that a specific protection system design basis be provided. The design basis should be reviewed to confirm that it has the following characteristics:

- Completeness - The design basis should address all system functions necessary to fulfill the system's safety intent. The design basis for protection systems should be shown to address the requirements of 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix A, "General Design Criteria for Nuclear Power Plants," General Design Criterion (GDC) 20, "Protection System Functions." Information provided for each design basis item should be sufficient to enable the detailed design of the I&C system to be carried out. All functional requirements for the I&C system and the operational environment for the I&C system should be described. As a minimum, each of the design basis aspects identified in IEEE Std 279-1971 Clauses 3(1) through 3(9) should be addressed.

- Consistency - The information provided in the design basis should be analyzed to confirm its consistency with the plant safety analysis, including the design basis event analysis of Chapter 15 of the SAR; the mechanical and electrical system designs; and other plant system designs.

The design bases should not contain contradictory requirements.

- Correctness - The information provided for the design basis items should be technically accurate.

- Traceability - It should be possible to trace the information in each design basis item back to the safety analyses, plant system design documents, regulatory requirements, applicant/licensee commitments, or other plant documents.

- Unambiguity - The information provided for the design basis items, taken alone and in combination, should have one and only one interpretation.

- Verifiability - The information provided for the design basis items should be stated or provided in such a way as to facilitate the establishment of verification criteria, and the performance of analyses and reviews of the various protection systems.

In addition to these characteristics, the following should be noted about the parts of IEEE Std 279-1971, Clause 3.

Clause 3(1) of IEEE Std 279-1971 requires in part the identification of conditions that require protective action. This information should be consistent with the analysis provided in Chapter 15 of the SAR. SRP Branch Technical Position (BTP) 7-4, "Guidance on Design Criteria for Auxiliary Feedwater Systems," provides specific guidance on the failures and malfunctions that should be considered in identification of design basis events for systems that initiate and control auxiliary feedwater systems. SRP BTP 7-5, "Guidance on Spurious Withdrawals of Single Control Rods in Pressurized Water Reactors," provides specific guidance on the reactivity control malfunctions that should be considered in the identification of conditions requiring protective action. The malfunctions assumed should be consistent with the control system failure modes described in Section 7.7 of the SAR and the reactivity control interlock functions described in Section 7.6 of the SAR.

Clause 3(2) of IEEE Std 279-1971 requires in part the identification of variables that are monitored in order to provide protective action. The tables in Sections 7.2 and 7.3 of the SAR should provide this information.

Clause 3(3) of IEEE Std 279-1971 requires in part the identification of the minimum number and location of sensors for those variables in Clause 3(2) of IEEE Std 279-1971 that have a spatial dependence. The applicant's or licensee's analysis should demonstrate that the number and location of sensors are adequate. Subsection 4.2 below discusses the consideration of the single failure criterion in the evaluation of this analysis.

Clauses 3(4), 3(5), and 3(6) of IEEE Std 279-1971 require in part the identification of operational limits, the margin between operational limits, and the level for the onset of unsafe conditions (setpoint), and limits that require protective action (safety limit - i.e., value assumed in the safety analysis) for each variable. The applicant's or licensee's analysis should confirm that an adequate margin exists between operating limits and setpoints, such that a low probability exists for inadvertent actuation of the system. The applicant's or licensee's analysis should confirm that an adequate margin exists between setpoints and safety limits, such that the system

initiates protective actions before safety limits are exceeded. RG 1.105, "Setpoints for Safety-Related Instrumentation," and BTP 7-12, "Guidance on Establishing and Maintaining Instrument Setpoints," provide guidance on the establishment of instrument setpoints. The instrument performance data used in setpoint analyses should be consistent with the performance requirements established in the design basis as discussed in Clause 3(9) of IEEE Std 279-1971. BTP 7-6, "Guidance on Design of Instrumentation and Controls Provided to Accomplish Changeover from Injection to Recirculation Mode," provides specific guidance for determining if the timing margins for changeover from injection to recirculation mode are sufficient to allow manual initiation of the transition.

Clause 3(7) of IEEE Std 279-1971 requires in part that the range of transient and steady-state conditions be identified for both the energy supply and the environment during normal, abnormal, and accident conditions under which the system must perform. This information is used in subsequent evaluations.

Clause 3(8) of IEEE Std 279-1971 requires in part the identification of malfunctions, accidents, or other unusual events that could physically damage protective system components or could cause environmental changes leading to functional degradation of system performance, and for which provisions must be incorporated to retain necessary protective action. This information is used in subsequent evaluations, with special attention given to Clause 4.4 of the standard, "Equipment Qualification."

Clause 3(9) of IEEE Std 279-1971 requires in part the identification of the minimum performance requirements including (a) system response times, (b) system accuracies, (c) ranges (normal, abnormal, and accident conditions) of the magnitudes, and rates of change of sensed variables to be accommodated until proper conclusion of the protective action is assured.

The applicant's or licensee's analysis, including the applicable portion provided in Chapter 15, should confirm that the system performance requirements are adequate to ensure completion of protective actions.

4. REQUIREMENTS

4.1. General Functional Requirements (IEEE Std 279-1971, Clause 4.1)

Clause 4.1 of IEEE Std 279-1971 requires in part that the protection system shall, with precision and reliability, automatically initiate protective action for the range of conditions and performance enumerated in Clauses 3(7) through 3(9) of IEEE Std 279-1971. The applicant's or licensee's analysis should confirm that the protection system has been qualified to demonstrate that the performance requirements are met. The evaluation should confirm that the general functional requirements have been appropriately allocated to the various system components. Automatic initiation is required for all protective functions; a manual initiation capability is also a requirement (see Clause 4.17 of IEEE Std 279-1971 and RG 1.62, "Manual Initiation of Protection Actions"). The evaluation of the precision of the protection system is addressed to the extent that setpoints, margins, errors, and response times are factored into the analysis. The topic of reliability is addressed in the following paragraphs.

Staff acceptance of system reliability is based on the deterministic criteria described in IEEE Std 279-1971 rather than on quantitative reliability goals. The NRC staff does not endorse the concept of quantitative reliability goals as a sole means of meeting the requirements for reliability of protection systems. Quantitative reliability determination, using a combination of analysis, testing, and operating experience can provide an added level of confidence in the reliable performance of the I&C system.

The applicant/licensee should justify that the degree of redundancy, diversity, testability, and quality provided in the protection system design is adequate to achieve functional reliability commensurate with the safety functions to be performed.

4.2. Single-Failure Criterion (IEEE Std 279-1971, Clause 4.2)

Clause 4.2 of IEEE Std 279-1971 requires in part that any single failure within the protection system shall not prevent proper protective action at the system level when required. The applicant's or licensee's analysis should confirm that the requirements of the single-failure criterion are satisfied. Guidance in the application of the single-failure criterion is provided in RG 1.53, "Application of the Single-Failure Criterion to Safety Systems," which endorses IEEE Std 379, "IEEE Standard Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems."

Where it is determined that the spatial dependence of a parameter requires several sensor channels to ensure plant protection, the redundancy requirements are determined for the individual case. In certain designs, for example, adequate monitoring of core power requires a minimum number of sensors arranged in a given configuration to provide adequate protection. This aspect of redundancy is dealt with in coordination with the organization responsible for the review of reactor systems to establish redundancy requirements.

Components and systems not qualified for seismic events or accident environments and nonsafety-grade components and systems are assumed to fail to function if failure adversely affects protection system performance. Conversely, these components and systems are assumed to function if functioning adversely affects protection system performance. All failures in the protection system that can be predicted as a result of an event for which the protection system is designed to provide a protective function are assumed to occur if the failure adversely affects the protection system performance. In general, the lack of equipment qualification may serve as a basis for the assumption of certain failures. After assuming the failures of nonsafety-grade, nonqualified equipment and those failures caused by a specific event, a random single failure is arbitrarily assumed. With these failures assumed, the protection system must be capable of performing the protective functions required to mitigate the consequences of the specific event.

4.3. Quality of Components and Modules (IEEE Std 279-1971, Clause 4.3)

The applicant or licensee should confirm that quality assurance provisions of Appendix B to 10 CFR Part 50, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," are applicable to the protection system. The evaluation of the adequacy of the quality assurance program is addressed in the review of Chapter 17 of the SAR.

4.4. **Equipment Qualification** (IEEE Std 279-1971 Clause 4.4)

The applicant or licensee should confirm that the protection system equipment is designed to meet the functional performance requirements over the range of environmental conditions for the area in which it is located, as identified by Clauses 3(7) and 3(9) of IEEE Std 279-1971, and discussed in Section 3 above.

I&C staff reviews mild environment qualification and electromagnetic interference (EMI) qualification of protection system I&C equipment, and consults with other organizations to confirm qualification for harsh environments and seismic loads.

RG 1.209, "Guidelines for Environmental Qualification of Safety-Related Computer-Based Instrumentation and Control Systems in Nuclear Power Plants," provides guidance for mild environment qualification. Additionally, the applicant or licensee should confirm that a single failure within the environmental control system, for any area in which protection system equipment is located, will not result in conditions that could result in damage to the protection system equipment, nor prevent the balance of the protection system not within the area from accomplishing its safety function. In this regard, the loss of an environmental control system is treated as a single failure that should not prevent the protection system from accomplishing its safety functions.

Because the loss of environmental control systems does not usually result in prompt changes in environmental conditions, the design bases may rely upon monitoring environmental conditions and taking appropriate action to ensure that extremes in environmental conditions are maintained within nondamage limits until the environmental control systems are returned to normal operation. If such bases are used, the applicant/licensee should confirm that there is independence between environmental control systems and sensing systems that would indicate the failure or malfunctioning of environmental control systems.

Review of mild environment qualification should also include confirmation that the environmental protection of instrument sensing lines conforms with the guidance of RG 1.151, "Instrument Sensing Lines."

EMI qualification in accordance with the guidance of RG 1.180, "Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems," is an acceptable means of meeting the qualification requirements for EMI and electrostatic discharge.

Lightning protection should be addressed as part of the review of electromagnetic compatibility. Lightning protection features should conform to the guidance of RG 1.204, "Guidelines for Lightning Protection of Nuclear Power Plants."

The organizations responsible for the review of equipment qualification to harsh environments and seismic events will perform the evaluation of conformance to the requirements of GDC 2 and 4 and 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants," to ensure the requirements for equipment qualification to harsh environments and seismic events are met. Guidance for the review of this equipment

qualification is given in SRP Sections 3.10, “Seismic and Dynamic Qualification of Mechanical and Electrical Equipment,” and 3.11, “Environmental Qualification of Mechanical and Electrical Equipment.”

4.5. Channel Integrity (IEEE Std 279-1971, Clause 4.5)

Information provided in Clauses 3(7) and 3(8) of IEEE Std 279-1971 is to be reviewed to confirm that the design includes the qualification of equipment for the conditions identified in the design bases. Failures may not be credited to protect the integrity of other equipment. The review should confirm that tests have been conducted on protection system equipment components and the system racks and panels as a whole to demonstrate the functional performance requirements of the protection system over the range of transient and steady-state conditions of both the energy supply and the environment. Where tests have not been conducted, the applicant should confirm that the protection system components are conservatively designed to operate over the range of service conditions.

Auxiliary features necessary to support protection system performance should meet all of the requirements of IEEE Std 279-1971. Other auxiliary features that are part of the protection system, but not isolated from the protection system, should be designed to meet the criteria of IEEE Std 279-1971 as necessary to assure that these components and systems do not degrade the protection systems below an acceptable level. SRP BTP 7-9, “Guidance on Requirements for Reactor Protection System Anticipatory Trips,” provides specific guidance for the review of anticipatory trips that are auxiliary features of a reactor protection system.

The sharing of structures, systems, and components between units in multi-unit stations is permissible provided that the ability to simultaneously perform required safety functions in all units is not impaired. The review of shared displays and controls should be coordinated with the organization responsible for the review of human factors to confirm that shared user interfaces are sufficient to support the operator needs for each of the shared units.

The organizations responsible for the review of electrical systems and balance of plant systems review power source requirements. Reviewers in the organization responsible for the review of I&Cs should coordinate with these organizations to confirm that I&C protection system power sources are adequate.

The review of channel integrity should confirm that the design provides for protection systems to fail in a safe state, or into a state that has been demonstrated to be acceptable on some other defined basis, if conditions such as disconnection of the system, loss of energy, or adverse environments are experienced. This aspect is typically evaluated through evaluation of the applicant’s or licensee’s failure modes and effects analysis. The analysis should justify the acceptability of each failure effect. The RTS functions should typically fail in the tripped state. ESFAS functions should fail to a predefined safe state. For many ESFAS functions this predefined safe state will be that the actuated component remains as-is.

4.6. Channel Independence (IEEE Std 279-1971, Clause 4.6)

Two aspects of independence should be addressed:

- Physical independence.
- Electrical independence.

Guidance for evaluation of physical and electrical channel independence is provided in RG 1.75, "Criteria for Independence of Electrical Safety Systems," which endorses IEEE Std 384, "IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits." The applicant or licensee should confirm that the protection system design precludes the use of components that are common to redundant channels, such as common switches for actuation, reset, mode, or test; common sensing lines; or any other features that could compromise the independence of redundant channels. Physical independence is attained by physical separation and physical barriers. Electrical independence shall include the utilization of separate power sources. The organization responsible for the review of electrical systems reviews power source requirements. Reviewers in the organization responsible for the review of I&Cs should coordinate with the electrical systems reviewers to confirm that I&C protection system power sources are adequate. Transmission of signals between independent channels should be through isolation devices.

SRP BTP 7-11, "Guidance on Application and Qualification of Isolation Devices," provides guidance for the application and qualification of isolation devices.

4.7. Control and Protection System Interaction (IEEE Std 279-1971, Clause 4.7)

Control and protection system interaction involves more than examining the electrical isolation and interconnection. The functional performance of control systems must be such that a control system cannot prevent proper action of a protection system. Clause 4.7 of IEEE Std 279-1971, with regard to isolation devices and multiple failures resulting from a credible single event, is explained by example in Clause 4.2 of IEEE Std 279-1971. The applicant's or licensee's analysis should confirm that the requirements for control and protection system interaction are satisfied.

4.8. Derivation of System Inputs (IEEE Std 279-1971, Clause 4.8)

A protection system that requires loss of flow protection would, for example, normally derive its signal from flow sensors. A design might use an indirect parameter such as a pressure signal or pump speed. However, the applicant/licensee should verify that any indirect parameter is a valid representation of the desired direct parameter for all events.

Even a directly measured variable should be reviewed and its response to postulated events compared with the credit taken for the parameter in the events for which it provides protection.

For both direct and indirect parameters, the applicant or licensee should verify that the characteristics (e.g., range, accuracy, resolution, response time) of the instruments that produce the protection system inputs are consistent with the analysis provided in Chapter 15 of the SAR.

4.9. Capability for Sensor Checks (IEEE Std 279-1971, Clause 4.9)

The most common method used to verify the availability of the input sensors is by cross checking between redundant channels that have available readout. When only two channels of readout are provided, the applicant or licensee should state the basis used to ensure that an operator will not take incorrect action when the two channel readouts differ. The applicant/licensee should state the method to be used for checking the operational availability of non-indicating sensors.

4.10. Capability for Test and Calibration (IEEE Std 279-1971, Clause 4.10)

Guidance on periodic testing of the protection system is provided in RG 1.22, "Periodic Testing of Protection System Actuation Functions," and in RG 1.118, "Periodic Testing of Electric Power and Protection Systems," which endorses IEEE Std 338, "Standard Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems." The extent of test and calibration capability provided bears heavily on whether the design meets the single-failure criterion. Any failure that is not detectable must be considered concurrently with any random postulated, detectable, single failure. Periodic testing should duplicate, as closely as practical, the overall performance required of the protection system. The test should confirm operability of both the automatic and manual circuitry. The capability should be provided to permit testing during power operation. When this capability can only be achieved by overlapping tests, the test scheme must be such that the tests do, in fact, overlap from one test segment to another. Test procedures that require disconnecting wires, installing jumpers, or other similar modifications of the installed equipment are not acceptable test procedures for use during power operation.

The review of test and calibration provisions should be coordinated with the organization responsible for the review of technical specification format and content to confirm that the system design supports the types of testing required by the technical specifications. The system design should also support the compensatory actions required by technical specifications when limiting conditions for operation are not met. Typically, the design should allow for tripping or bypass of individual functions in each protection system channel.

4.11. Channel Bypass and Removal from Operation (IEEE Std 279-1971, Clause 4.11)

The review of bypass and removal from operations should be coordinated with the organization that is responsible for the format of technical specifications to confirm that the provisions for this bypass are consistent with the required actions of the proposed plant technical specifications.

4.12. Operating Bypass (IEEE Std 279-1971, Clause 4.12)

The requirement for automatic removal of operational bypasses means that the reactor operator shall have no role in such removal. The operator may take action to prevent the unnecessary initiation of a protective action.

4.13. Indication of Bypass (IEEE Std 279-1971, Clause 4.13)

Guidance on bypasses and inoperable status indication is provided in RG 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety System."

4.14. Access to Means for Bypassing (IEEE Std 279-1971, Clause 4.14)

Administrative control is acceptable to ensure that access to the means for bypassing is limited to qualified plant personnel and that permission of the control room operator is obtained to gain access.

4.15. Multiple Setpoints (IEEE Std 279-1971, Clause 4.15)

The staff interpretation of "positive means" is that automatic action is provided to ensure that the more restrictive setpoint is used when required. SRP BTP 7-3, "Guidance on Protection System Trip Point Changes for Operation with Reactor Coolant Pumps Out of Service," provides additional guidance on multiple setpoints used to allow operation with reactor coolant pumps out of service.

4.16. Completion of a Protective Action Once it is Initiated
(IEEE Std 279-1971, Clause 4.16)

The staff review of this item should include review of functional and logic diagrams to ensure that "seal-in" features are provided to enable system-level protective actions to go to completion. The seal-in feature may incorporate a time delay as appropriate for the safety function. Additionally, the seal-in feature need not function until it is confirmed that a valid protective command has been received, provided the system meets response time requirements.

4.17. Manual Initiation (IEEE Std 279-1971, Clause 4.17)

Features for manual initiation of protective action should conform with RG 1.62, "Manual Initiation of Protection Actions."

The review of manual controls should be coordinated with the organization responsible for the review of human factors to confirm that the functions controlled and the characteristics of the controls (e.g., location, range, type, and resolution) allow plant operators to take appropriate manual actions.

The review of manual controls should include confirmation that the controls will be functional (e.g., power will be available and command equipment is appropriately qualified) during plant conditions under which manual actions may be necessary.

4.18. Access to Setpoint Adjustments, Calibrations, and Test Points
(IEEE Std 279-1971, Clause 4.18)

The review of access control should confirm that design features provide the means to control physical access to protection system equipment, including access to test points and means for changing setpoints. Typically such access control includes provisions such as alarms and locks on protection system panel doors, or control of access to rooms in which protection system equipment is located.

4.19. Identification of Protective Actions and Information Read-Out
(IEEE Std 279-1971, Clauses 4.19 and 4.20)

The review of information displays should be coordinated with the organization that is responsible for the review of reactor systems to confirm that the information displayed and characteristics of the displays (e.g., location, range, type, and resolution) support operator awareness of system and plant status and will allow plant operators to make appropriate decisions.

The review of information displays for manually controlled actions should include confirmation that displays will be functional (e.g., power will be available and sensors are appropriately qualified) during plant conditions under which manual actions may be necessary.

Protection system bypass and inoperable status indication should conform with the guidance of RG 1.47.

4.20. Information Read-Out (IEEE Std 279-1971, Clause 4.20)

See Subsection 4.19 above.

4.21. System Repair (IEEE Std 279-1971, Clause 4.21)

Protection systems may include self-diagnostic capabilities to aid in troubleshooting.

4.22. Identification (IEEE Std 279-1971, Clause 4.22)

Guidance on identification is provided in RG 1.75, which endorses IEEE Std 384. The preferred identification method is color coding of components, cables, and cabinets.

5. REFERENCES

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

2. Institute of Electrical and Electronics Engineers, IEEE Std 323-1974, “IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations,” Piscataway, NJ.

3. Institute of Electrical and Electronics Engineers, IEEE Std 338, “Standard Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems,” Piscataway, NJ.

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

5. Institute of Electrical and Electronics Engineers, IEEE Std 384, “IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits,” Piscataway, NJ.

6. Institute of Electrical and Electronics Engineers, IEEE Std 603-1991, “IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations,” Piscataway, NJ.

7. Institute of Electrical and Electronics Engineers, IEEE Std 665, “Guide for Generation Station Grounding,” Piscataway, NJ.

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

9. U.S. Nuclear Regulatory Commission, “Periodic Testing of Protection System Actuation Functions,” Regulatory Guide 1.22.

10. U.S. Nuclear Regulatory Commission, “Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety System,” Regulatory Guide 1.47.

11. U.S. Nuclear Regulatory Commission, “Application of the Single-Failure Criterion to Safety Systems,” Regulatory Guide 1.53.

12. U.S. Nuclear Regulatory Commission, “Manual Initiation of Protection Action,” Regulatory Guide 1.62.

13. U.S. Nuclear Regulatory Commission, “Criteria for Independence of Electrical Safety Systems,” Regulatory Guide 1.75.

14. U.S. Nuclear Regulatory Commission, “Setpoints for Safety-Related Instrumentation,” Regulatory Guide 1.105.

15. U.S. Nuclear Regulatory Commission, “Periodic Testing of Electric Power and Protection Systems,” Regulatory Guide 1.118.

16. U.S. Nuclear Regulatory Commission, “Instrument Sensing Lines,” Regulatory Guide 1.151.

17. U.S. Nuclear Regulatory Commission, “Criteria for Use of Computers in Safety Systems of Nuclear Power Plants,” Regulatory Guide 1.152.

18. U.S. Nuclear Regulatory Commission, “Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems,” Regulatory Guide 1.180.

19. U.S. Nuclear Regulatory Commission, “Guidelines for Lightning Protection of Nuclear Power Plants,” Regulatory Guide 1.204.

20. U.S. Nuclear Regulatory Commission, “Guidelines for Environmental Qualification of Safety-Related Computer-Based Instrumentation and Control Systems in Nuclear Power Plants,” Regulatory Guide 1.209.

21. U.S. Nuclear Regulatory Commission, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs,” SECY-93-087, April 2, 1993.

22. U.S. Nuclear Regulatory Commission, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs,” Staff Requirements Memorandum on SECY-93-087, July 15, 1993.

PAPERWORK REDUCTION ACT STATEMENT

The information collections contained in the Standard Review Plan are covered by the requirements of 10 CFR 50, and was approved by the Office of Management and Budget, approval number 3150-0011.

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.

APPENDIX 7.1-B
Description of Changes

APPENDIX 7.1-B, "Guidance for Evaluation of Conformance to IEEE Std 279"

This Appendix 7.1-B Section affirms the technical accuracy and adequacy of the guidance previously provided in Appendix 7.1-B, Revision 5, dated March 2007. See ADAMS Accession No. ML070550087.

The main purpose of this update is to incorporate the revised software Regulatory Guides and the associated endorsed standards. For organizational purposes, the revision number of each Regulatory Guide and year of each endorsed standard is now listed in one place, Table 7-1. As a result, revisions of Regulatory Guides and years of endorsed standards were removed from this section, if applicable. For standards that are incorporated by reference into regulation (IEEE Std 279-1971 and IEEE Std 603-1991) and standards that have not been endorsed by the agency, the associated revision number or year is still listed in the discussion.

Added Regulatory Guide 1.209, "Guidelines for Environmental Qualification of Safety Related Computer-Based Instrumentation and Control Systems in Nuclear Power Plants." to the list of applicable regulatory guides for reviews under this SRP section, and to the discussion of equipment qualification for mild environments.

Part of 10 CFR was reorganized due to a rulemaking in the fall of 2014. Quality requirement discussions in the former 10 CFR 50.55a(a)(1) were moved to 10 CFR 50.54(jj) and 10 CFR 50.55(i). The incorporation by reference language in the former 10 CFR 50.55a(h)(1) was moved to 10 CFR 50.55a(a)(2). There were no changes either to 10 CFR 50.55a(h)(2) or 10 CFR 50.55a(h)(3).

Additional changes were editorial.