



NUREG-0800

U.S. NUCLEAR REGULATORY COMMISSION STANDARD REVIEW PLAN

7.3 ENGINEERED SAFETY FEATURES SYSTEMS

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 (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.

I. AREAS OF REVIEW

Draft Revision 6 - August 2015

USNRC STANDARD REVIEW PLAN

This Standard Review Plan (SRP), NUREG-0800, has been prepared to establish criteria that the U.S. Nuclear Regulatory Commission (NRC) staff responsible for the review of applications to construct and operate nuclear power plants intends to use in evaluating whether an applicant/licensee meets the NRC regulations. The SRP is not a substitute for the NRC regulations, and compliance with it is not required. However, an applicant is required to identify differences between the design features, analytical techniques, and procedural measures proposed for its facility and the SRP acceptance criteria and evaluate how the proposed alternatives to the SRP acceptance criteria provide an acceptable method of complying with the NRC regulations.

The SRP sections are numbered in accordance with corresponding sections in Regulatory Guide (RG) 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)." Not all sections of RG 1.70 have a corresponding review plan section. The SRP sections applicable to a combined license application for a new light-water reactor (LWR) are based on RG 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)."

These documents are made available to the public as part of the NRC policy to inform the nuclear industry and the general public of regulatory procedures and policies. Individual sections of NUREG-0800 will be revised periodically, as appropriate, to accommodate comments and to reflect new information and experience. Comments may be submitted electronically by e-mail to NRO_SRP.Resource@nrc.gov.

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The objective of the review is to confirm that the engineered safety features actuation system (ESFAS) and engineered safety features (ESF) control systems satisfy regulatory acceptance ~~criteria, guidelines and performance requirements.~~

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Draft Revision 6 - August 2015

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criteria, guidelines and performance requirements.

1. This ~~Standard Review Plan (SRP)~~ section describes the review process and acceptance criteria for the ESFAS, which is a portion of the protection system used to initiate the ESF systems and auxiliary supporting features and other auxiliary features. The ESFAS provides both automatic and manual initiation of these systems. This SRP section also includes the review criteria for control systems that regulate the ESF systems. The ESF control systems include both the automatic and manual features.

Revision 5 – March 2007

~~USNRC STANDARD REVIEW PLAN~~

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~~The specific areas of review are as follows:~~

~~The review of instrumentation and control systems that regulate the operation of auxiliary supporting features and other auxiliary features is included in the SRP sections that address systems that implement those features. SRP Section 7.5, "Information Systems Important to Safety," provides the review criteria for the information systems important to safety, which ~~includes~~ instrumentation ~~that~~ indicates the need for manual initiation and control of ESF systems. Examples of ESF systems, Auxiliary Supporting Features and Other Auxiliary Features are as follows:~~

~~Typical ESF systems are:~~

- ~~A. A. Containment and reactor vessel isolation systems.~~
- ~~B. B. Emergency core cooling systems.~~
- ~~C. C. Containment heat removal and depressurization systems.~~
- ~~D. D. Pressurized water reactor auxiliary feedwater systems.~~
- ~~E. E. Emergency boration systems.~~

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F. Boiling water reactor standby gas treatment systems.

G. Containment air purification and cleanup systems.

H. Containment combustible gas control systems.

I. Control room isolation and emergency heating, ventilating, and air conditioning (HVAC).

Typical auxiliary supporting features and other auxiliary features (additional information is provided in Appendix 7.1-C) are: A. Electric power systems.

A. Electric power systems

B. Diesel generator fuel storage and transfer systems.

C. Instrument air systems.

D. HVAC systems for ESF areas.

E. Essential service water and component cooling water systems.

Figure 3 of IEEE Std 603-1991, "Examples of Equipment Fitted to Safety Systems Scope Diagram," of the Institute of Electrical and Electronics Engineers (IEEE) Standard (Std) 603-1991, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations," provides a matrix with an extensive list of auxiliary supporting features and other auxiliary features.

2. Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC). For design certification (DC) and combined license (COL) reviews, the staff reviews the applicant's proposed

ITAAC associated with the structures, systems, and components (SSCs) related to this SRP section in accordance with SRP Section 14.3, "Inspections, Tests, Analyses, and Acceptance Criteria." The staff recognizes that the review of ITAAC cannot be completed until after the rest of this portion of the application has been reviewed against acceptance criteria contained in this SRP section. Furthermore, the staff reviews the ITAAC to ensure that all SSCs in this area of review are identified and addressed as appropriate in accordance with SRP Section- 14.3.

3. COL Action Items and Certification Requirements and Restrictions. For a DC application, the review will also address COL action items and requirements and restrictions (e.g., interface requirements and site parameters).

For a COL application referencing a DC, a COL applicant must address COL action items (referred to as COL license information in certain DCs) included in the referenced

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DC. Additionally, a COL applicant must address requirements and restrictions (e.g., interface requirements and site parameters) included in the referenced DC.

Review Interfaces

Other SRP sections interface with this section as follows:

1. SRP Section 7.0, "Instrumentation and Controls - Overview of Review Process," describes the coordination of reviews, including the information to be reviewed and the scope required for each of the different types of applications that the staff may review. Refer to that section for information regarding how the areas of review are affected by the type of application under consideration and for a description of coordination between the organization responsible for the review of instrumentation and control systems and other organizations.

The specific acceptance criteria and review procedures are contained in the referenced SRP sections.

II. ACCEPTANCE CRITERIA

Requirements

Acceptance criteria are based on meeting the relevant requirements of the following Commission regulations:

Acceptance criteria applicable to any ESFAS and ESF control systems:

1. 10 CFR 50.55a(a)(1), "Quality Standards."

1. Title 10 of the Code of Federal Regulations (10 CFR) 50.54(jj) and 10 CFR 50.55(i).

2. 10 CFR 50.55a(h), "Protection and Safety Systems," which requires compliance with IEEE

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 before January 1, 1971, the applicant or licensee may elect to comply instead with their plant specific licensing basis.

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2. For nuclear power plants with construction permits issued between January 1, 1971, and May 13, 1999, the applicant or licensee may elect to comply instead with the requirements stated in IEEE Std 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations."

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3. 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix A, A, "General Design Criteria for Nuclear Plants," General Design Criterion (GDC)- 1, "Quality Standards and Records."

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4. GDC 2, "Design Basis for Protection Against Natural Phenomena."

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5. GDC 4, "Environmental and Dynamic Effects, Missile Design Basis."

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6. GDC 10, "Reactor Design."

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7. GDC 13, "Instrumentation and Control."

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8. GDC 15, "Reactor Coolant System Design."

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9. GDC 16, "Containment Design."

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10. GDC 19, "Control Room."

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11. 10 CFR 52.47(b)(1), which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's U.S. Nuclear Regulatory Commission's (NRC's) regulations.

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12. 10 CFR 52.80(a), which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.

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Additional acceptance criteria applicable to the ESFAS:

1. 10 CFR 50.34(f) "Additional TMI- Related Requirements," or equivalent Three Mile Island (TMI) action requirements imposed by Generic Letters: (GL),

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10 CFR 50.34(f)(2)(v), "Bypass and Inoperable Status Indication," and 10 CFR 50.34(f)(2)(xii), "Auxiliary Feedwater System Automatic Initiation and Flow Indication," and

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(10 CFR 50.34(f)(2)(xiv), "Containment Isolation Systems."

2. GDC 20, "Protection ~~Systems Function~~System Functions."

3. GDC 21, "Protection System Reliability and Testability."

4. GDC 22, "Protective System Independence."

5. GDC 23, "Protection System Failure Modes."

6. GDC 24, "Separation of Protection and Control Systems."

7. GDC 29, "Protection against Anticipated Operational Occurrences."

Additional acceptance criteria applicable to ESF control systems:

1. GDC 33, "Reactor Coolant Makeup."

2. GDC 34, "Residual Heat Removal."

3. GDC 35, "Emergency Core Cooling."

4. GDC 38, "Containment Heat Removal."

5. GDC 41, "Containment Atmosphere Cleanup."

6. GDC 44, "Cooling Water."

SRP Acceptance Criteria

Specific SRP acceptance criteria acceptable to meet the relevant requirements of the NRC's regulations identified above are contained in SRP Section 7.1, SRP Table 7-1 and SRP Appendix 7.1-A which list standards, ~~regulatory guides~~RGs, and branch technical positions (BTPs). -The SRP is not a substitute for the NRC's regulations, and compliance with it is not required. However, an applicant is required to identify differences between the design features, analytical techniques, and procedural measures proposed for its facility and the SRP acceptance criteria and evaluate how the proposed alternatives to the SRP acceptance criteria provide acceptable methods of compliance with the NRC regulations.

1. SRP Appendix 7.1-C provides SRP acceptance criteria for safety system compliance with 10 CFR 50.55a(h).

2. SRP Appendix 7.1-B provides SRP acceptance criteria for protection system compliance with 10 CFR 50.55a(h).

3. ~~IEEE Std 7-4.3.2-2003~~, "IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations," as endorsed by ~~Regulatory Guide~~RG 1.152; ~~Revision 2~~, "Criteria for Digital Computers in Safety Systems of Nuclear Power Plants,"

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provides guidance on applying the safety system criteria to computer-based safety systems. SRP Appendix 7.1-D provides SRP acceptance criteria for safety and protection systems using digital computer-based technology.

- 4. Item II.Q, "Defense Against Common-Mode Failures in Digital Instrument and Control Systems," of the Staff Requirements Memorandum on SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light Water Reactor (ALWR) Designs," provides guidance on Diversity and Defense-in-Depth. SRP BTP 7-19 provides additional guidance.

III. REVIEW PROCEDURES

The reviewer will select material from the procedures described below, as may be appropriate for a particular case. Typical reasons for a non-uniform emphasis are the introduction of new design features or the utilization in the design of features previously reviewed and found acceptable.

These review procedures are based on the identified SRP acceptance criteria. For deviations from these specific acceptance criteria, the staff should review the applicant's evaluation of how the proposed alternatives to the SRP criteria provide an acceptable method of complying with the relevant NRC requirements identified in Subsection II.

SRP Section 7.1 describes the general procedures to be followed in reviewing any instrumentation and control system. This part of SRP Section 7.3 highlights specific topics that should be emphasized in the ESFAS review.

- 1. The review should include an evaluation of the ESFAS design against the requirements of IEEE Std 603-1991 or IEEE Std 279-1971, depending upon the applicant's or licensee's commitment regarding these design criteria. For computer-based ESFAS, guidance is provided by IEEE Std 7-4.3.2-2003 as endorsed by Regulatory Guide 1.152, Revision as endorsed by RG 1.152. These procedures are detailed in SRP Appendix 7.1-B for IEEE Std 279-1971, SRP Appendix 7.1-C for IEEE Std 603-1991, and SRP Appendix 7.1-D for IEEE Std 7-4.3.2-2003. These procedures are detailed in SRP Appendix 7.1 B for IEEE Std 279-1971, SRP Appendix 7.1 C for IEEE Std 603-1991, and SRP Appendix 7.1 D for IEEE Std 7-4.3.2-2003.

SRP Appendices 7.1-B and 7.1-C discuss the requirements of IEEE Std 279-1971 and

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IEEE Std 603-1991 and how they are used in the review of the ESFAS. SRP Appendix 7.1-D discusses the criteria of IEEE Std 7-4.3.2-2003, and how they are used in the review of the ESFAS. Although the primary emphasis is on the equipment comprising the ESFAS, the reviewer must consider the overall ESF functions at the system level. The ESFAS design should be compatible with the accident analyses. It is not sufficient to evaluate the adequacy of the ESFAS only on the basis of the design meeting the specific requirements of IEEE Std 279-1971 or IEEE Std 603-1991.

The ESFAS review should address the applicable topics identified as applicable in SRP Table 7-1. SRP Appendix 7.1-A describes review methods for each topic. Major design considerations that should be emphasized in the ESFAS review are identified below.

- A. Design basis — See SRP Appendix 7.1-B-subsection, Subsection 3 or SRP Appendix 7.1-C-subsection, Subsection 4.
- B. Single-failure criterion — See SRP Appendix 7.1-B-subsection, Subsection 4.2 or SRP Appendix 7.1-C-subsection, Subsection 5.1.
- C. Quality of components and modules — See SRP Appendix 7.1-B-subsection, Subsection 4.3 or SRP Appendix 7.1-C-subsection, Subsection 5.3.
- D. Independence — See SRP Appendix 7.1-B-subsections, Subsections 4.6 and 4.7 or SRP Appendix 7.1-C-subsections, Subsections 5.6 and 6.3.
- E. Completion of protective action — See SRP Appendix 7.1-B-subsection, Subsection 4.16 or SRP Appendix 7.1-C-subsection, Subsection 5.2.
- F. Diversity and defense-in-depth — ESFAS should incorporate multiple means for responding to each event discussed in the SAR Chapter 15, “Transient and Accident Analyses.” At least one pair of these means for each event should have the property of signal diversity, i.e., the use of different sensed parameters to initiate protective action, in which any of the parameters may independently indicate an abnormal condition, even if the other parameters are sensed incorrectly (see NUREG/CR-6303, “Method for Performing Diversity and Defense-in-Depth Analyses of Reactor Protection Systems”). The diverse means may actuate the same protective function or different protective functions, and may be automatically or manually activated, consistent with the response time requirements of the function. For digital computer-based ESFAS systems, the applicant or licensee should have performed a defense-in-depth and diversity analysis. Additionally, for advanced reactor design under 10 CFR Part 52, “Licenses, Certifications, and Approvals for Nuclear Power Plants,” the design should provide for manual, system-level actuation of critical safety functions. SRP BTP 7-19 provides guidance for the review of diversity and defense-in-depth.

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G. System testing and inoperable surveillance — See SRP Appendix SRP 7.1-B subsections, Subsections 4.9 and 4.10 or SRP Appendix 7.1-C subsections, Subsections 5.7, 5.8, and 6.5. H. —

G.H. Use of digital systems — See SRP Appendix 7.0-A and SRP Appendix 7.1-D.

I. Setpoint determination — See Regulatory Guide RG 1.105, “Setpoints for Safety-Related Instrumentation,” and SRP BTP 7-12, BTP 7-3, Appendix 7.1-B subsection, Subsection 4.18, and Appendix 7.1-C subsection, Subsection 6.8.

J. ESF control systems — Conformance to the single-failure criterion on a system basis, and operability from onsite and offsite electrical power as required by GDC General Design Criteria 34, 35, 38, and 41.

2. For review of a DC application, the reviewer should follow the above procedures to verify that the design, including requirements and restrictions (e.g., interface requirements and site parameters), set forth in the final safety analysis report (FSAR) meets the acceptance criteria. DCs have referred to the FSAR as the design control document (DCD). The reviewer should also consider the appropriateness of identified COL action items. The reviewer may identify additional COL action items; however, to ensure these COL action items are addressed during a COL application, they should be added to the DC FSAR.

For review of a COL application, the scope of the review is dependent on whether the COL applicant references a DC, an early site permit (ESP) or other NRC approvals (e.g., manufacturing license, site suitability report or topical report).

3. For review of both DC and COL applications, SRP Section 14.3 should be followed for the review of ITAAC. The review of ITAAC cannot be completed until after the completion of this section.

IV. EVALUATION FINDINGS

The reviewer verifies that the applicant has provided sufficient information and that the review and calculations (if applicable) support conclusions of the following type to be included in the staff's safety evaluation report (SER). The reviewer also states the bases for those conclusions.

1. The review of the instrumentation and control aspects of the engineered safety feature (ESF) systems includes the engineered safety features actuation systems (ESFAS) and the ESF control systems. The ESFAS detects a plant condition requiring the operation of an ESF system and/or auxiliary supporting features and other auxiliary features and initiates operation of the systems. The ESF control systems regulate the operation of the ESF systems following automatic initiation by the protection system or manual initiation by the plant operator.

The NRC staff concludes that the design of the ESFAS is acceptable and meets the relevant requirements of General Design Criteria (GDC) 1, 2, 4, 10, 13, 15, 16, 19-24,

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29, 33, 34, 35, 38, 41, and ~~44~~, and 10 CFR 50.34(f), ~~10 CFR 50.55a(a)(1), 54(jj)~~ and 10 CFR 50.55(i), and ~~10 CFR 50.55a(h)~~.

The staff conducted a review of these systems for conformance to the guidelines in the ~~regulatory guides~~RGs, industry standards and branch technical positions applicable to these systems. The staff concludes that the applicant/licensee acceptably identified the guidelines applicable to these systems. Based upon the review of the system design for conformance to the guidelines, the staff concludes that the systems conform to the guidelines applicable to these systems. Therefore, the staff finds that the requirements of GDC 1 and ~~10 CFR 50.55a(a)(1), 54(jj)~~ and 10 CFR 50.55(i) have been met.

The review included the identification of those systems and components for the ESFAS and ESF control systems that are designed to survive the effects of earthquakes, other natural phenomena, abnormal environments, and missiles. Based upon the review, the staff concludes that the applicant ~~or~~ licensee has identified those systems and components consistent with the design bases for those systems. Sections 3.10 and 3.11 of the SER address the qualification programs to demonstrate the capability of these systems and components to survive the above effects. Therefore, the staff finds that the identification of these systems and components satisfies the requirements of ~~GDC~~General Design Criteria 2 and 4.

Based on the review of ESFAS and ESF control system status information, manual initiation capabilities, control capabilities, and provisions to support safe shutdown, the staff concludes that information is provided to monitor the system over the anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety. Appropriate controls are provided for manual initiation and control of ESF functions. ESF controls appropriately support actions to operate the nuclear power unit safely under normal conditions and to achieve and maintain a safe condition under accident conditions. Therefore, the staff finds that the ESFAS and ESF control design satisfies the requirements of ~~GDC~~General Design Criteria 13 and 19.

Based on the review of system functions, the staff concludes that the ESFAS ~~conforms conform~~ to the requirements of ~~(IEEE Std 279-1971 or IEEE Std 603-1991)~~ and 10 CFR 50.34(f). The ESFAS setpoint methodology conforms to the guidance of ~~Regulatory Guide~~RG 1.105, "Setpoints for Safety-Related Instrumentation." Based upon this review and coordination with those having primary review responsibility for the accident analysis, the staff concludes that the ESFAS includes the provision to sense accident conditions and anticipated operational occurrences consistent with the accident analysis presented in Chapter 15 of the SAR and evaluated in the SER. Therefore, the staff finds that the ESFAS satisfies the requirements of GDC 20.

The ESFAS conforms to the guidelines for periodic testing in ~~Regulatory Guide~~RG 1.22, "Periodic Testing of Protection System Actuation Functions," and ~~Regulatory Guide~~RG 1.118, "Periodic Testing of Electric Power and Protection Systems." The bypassed and inoperable status indication conforms to the guidelines of ~~Regulatory Guide~~RG 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety

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Systems.” The ESFAS conforms to the guidelines on the application of the single-failure criterion in IEEE Std 379-2000, “IEEE Standard Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems,” as supplemented by Regulatory Guide RG 1.53, “Application of the Single-Failure Criterion to Safety Systems.”

Based on the review, the staff concludes that the ESFAS satisfies the requirement of (IEEE Std 279-1971 or IEEE Std 603-1991) with regard to the system reliability and testability. Therefore, the staff finds that the ESFAS satisfies these requirements of GDC-21.

The ESFAS conforms to the guidelines in Regulatory Guide RG 1.75, “Criteria for Independence of Electrical Safety Systems,” for the protection system independence. Based on the review, the staff concludes that the ESFAS satisfies the requirement of (IEEE Std 279-1971 or IEEE Std 603-1991) with regard to the system’s independence. Therefore, the staff finds that the ESFAS satisfies the requirements of GDC 22.

Based on the review of the failure modes and effects analysis for the ESFAS, the staff concludes that the system is designed to fail into a safe state if conditions such as disconnection of the system, loss of energy, or a postulated adverse environment are experienced. Therefore, the staff finds that the ESFAS satisfies the requirements of GDC-23.

Based on the review of the interfaces between the ESFAS and plant operating control systems, the staff concludes that the system satisfies the requirements of (IEEE Std 279-1971 or IEEE Std 603-1991) with regard to control and protection system interactions. Therefore, the staff finds the ESFAS satisfies the requirements of GDC 24.

Based on the review of the ESFAS, the staff concludes that the system satisfies the protection system requirements for malfunctions of the reactivity control system such as accidental withdrawal of control rods. Chapter 15 of the SAR and SER address the capability of the system to ensure that fuel design limits are not exceeded for such events. Therefore, the staff finds that the RTS reactor trip system satisfies the requirements of GDC 25.

The staff conducted a review of the ESF control systems for conformance to the requirements for testability, operability with onsite and offsite electrical power, and single failures. The staff concludes that the ESF control systems are testable and are operable using either onsite or offsite power (assuming only one source is available). Additionally, the controls associated with redundant ESF systems are independent and satisfy the single-failure criterion and, therefore, meet the relevant requirements of GDC General Design Criteria 34, 35, 38, and 41.

The conclusions noted above are based upon the requirements of (IEEE Std 279-1971 or IEEE Std 603-1991) with respect to the design of the ESFAS. Therefore, the staff finds that the ESFAS satisfies the requirements of 10 CFR 50.55a(h).

The applicant or licensee has also incorporated in the system design the

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f (recommendations of the TMI task action plan items ~~OR~~ the requirements of 10-CFR 50.34(f), f), (identify item number and how implemented,) which the staff has reviewed and found acceptable.

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In the review of the ESFAS, the staff examined the dependence of this system on the availability of essential auxiliary systems. Based on this review and coordination with those having primary review responsibility of auxiliary supporting features and other auxiliary features systems, the staff concludes that the design of the ESFAS is compatible with the functional requirements of auxiliary supporting features and other auxiliary features systems.

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2. Note: The following finding applies to systems involving computer-based components.

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Based on the review of software development plans and the review of the computer software development process and design outputs, the staff concludes that the computer systems meet the guidance of ~~Regulatory Guide~~RG 1.152. Therefore, the special characteristics of computer systems have been adequately addressed, and the staff finds that the ESFAS satisfies these requirements of ~~GDC~~General Design Criteria 1 and 21.

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Based on the review of the applicant/licensee's defense-in-depth and diversity analysis, the staff concludes that the ESFAS complies with the criteria for defense against common-cause failure in digital instrumentation and control systems. Therefore, the staff finds that adequate diversity and defense against common-cause failure have been provided to satisfy these requirements of ~~GDC~~General Design Criteria 21 and 22, and the ~~Staff Requirements Memorandum~~staff requirements memorandum on SECY-93-087.

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3. For DC and COL reviews, the findings will also summarize the staff's evaluation of requirements and restrictions (e.g., interface requirements and site parameters) and COL action items relevant to this SRP section.

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4. Note: The following conclusion is applicable to all applications.

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The conclusions noted above for the ESFAS are applicable to all portions of the systems except for the following, for which acceptance is based upon prior NRC review and approval as noted [list applicable system or topics and identify references].

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5. In addition, to the extent that the review is not discussed in other SER sections, the findings will summarize the ~~staff's~~staff's evaluation of the ITAAC, including design acceptance criteria, as applicable.

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V. IMPLEMENTATION

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The staff will use this SRP section in performing safety evaluations of DC applications and license applications submitted by applicants pursuant to 10-CFR Part 50 or 10 CFR Part 52. Except when the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the staff will use the method described herein to evaluate conformance with Commission regulations.

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The provisions of this SRP section apply to reviews of applications submitted ~~six~~6 months or more after the date of issuance of this SRP section, unless superseded by a later revision.

VI. REFERENCES

1. IEEE Std 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations."
2. IEEE Std 379-2000, "IEEE Standard Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems."
3. IEEE Std 603-1991, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations."
4. IEEE Std 7-4.3.2-2003, "IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations."
5. NUREG/CR-6303, "Method for Performing Diversity and Defense-in-Depth Analyses of Reactor Protection Systems," December 1994.
Regulatory Guide
6. RG 1.105, "Setpoints for Safety-related Related Instrumentation." Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, 1999.
Regulatory Guide
7. RG 1.118, "Periodic Testing of Electric Power and Protection Systems." Office of Nuclear Regulatory Research, U.S. Nuclear regulatory Commission, 1995.
Regulatory Guide
8. RG 1.152, "Criteria for Digital Use of Computers in Safety Systems of Nuclear Power Plants." Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, 2006.
Regulatory Guide
9. RG 1.22, "Periodic Testing of Protection System Actuation Functions." Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, 1972.
Regulatory Guide
10. RG 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems." Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, 1973.
Regulatory Guide
11. RG 1.53, "Application of the Single-Failure Criterion to Safety Systems." Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, 2003.
Regulatory Guide
12. RG 1.75, "Criteria for Independence of Electrical Safety Systems." Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, 2005.
13. SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light Water Reactor (ALWR) Designs," April 2, 1993.

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14. Staff Requirements Memorandum on SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light Water Reactor (ALWR) Designs," July 1993.

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PAPERWORK REDUCTION ACT STATEMENT

The information collections contained in the Standard Review Plan are covered by the requirements of 10 CFR Part 50 and 10- CFR- Part 52, and were approved by the Office of Management and Budget, approval number 3150-0011 and 3150-0151.

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.

Revision 6 - xxxxx

USNRC STANDARD REVIEW PLAN

This Standard Review Plan, NUREG-0800, has been prepared to establish criteria that the U.S. Nuclear Regulatory Commission staff responsible for the review of applications to construct and operate nuclear power plants intends to use in evaluating whether an applicant/licensee meets the NRC's regulations. The Standard Review Plan is not a substitute for the NRC's regulations, and compliance with it is not required. However, an applicant is required to identify differences between the design features, analytical techniques, and procedural measures proposed for its facility and the SRP acceptance criteria and evaluate how the proposed alternatives to the SRP acceptance criteria provide an acceptable method of complying with the NRC regulations.

The standard review plan sections are numbered in accordance with corresponding sections in Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)." Not all sections of Regulatory Guide 1.70 have a corresponding review plan section. The SRP sections applicable to a combined license application for a new light-water reactor (LWR) are based on Regulatory Guide 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)."

These documents are made available to the public as part of the NRC's policy to inform the nuclear industry and the general public of regulatory procedures and policies. Individual sections of NUREG-0800 will be revised periodically, as appropriate, to accommodate comments and to reflect new information and experience. Comments may be submitted electronically by email to NRR_SRP@nrc.gov.

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**SRP Section 7.3
Description of Changes**

SRP Section 7.3, "Engineered Safety Features Systems"

This SRP Section affirms the technical accuracy and adequacy of the guidance previously provided in SRP Section 7.3, Revision 5, dated March 2007. See ADAMS Accession Number ML070560004.

The main purpose of this update is to incorporate the revised software RGs 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. Additional changes were editorial.

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).

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