U.S. NUCLEAR REGULATORY COMMISSION

DESIGN-SPECIFIC REVIEW STANDARD for NuScale SMR DESIGN

7.0 INSTRUMENTATION AND CONTROLS—INTRODUCTION AND OVERVIEW OF REVIEW PROCESS

This Design-Specific Review Standard (DSRS) section provides guidance to the staff of the U.S. Nuclear Regulatory Commission (NRC) to use in reviewing the instrumentation and control (I&C) design of the NuScale Power (NuScale) small modular reactor (SMR) nuclear power reactor. This guidance will help the staff in determining whether the design complies with the applicable regulatory requirements and whether the applicant has demonstrated that there is reasonable assurance that the design will adequately protect public health and safety. This DSRS was developed as a pilot initiative for the NuScale SMR design, and is not applicable to other designs unless specifically addressed in DSRS documents for that design center because this guidance focuses on NuScale SMR design-specific technical matters.

Major Differences between the DSRS and the Standard Review Plan

The guidance in this DSRS chapter differs from the guidance in Chapter 7 of the Standard Review Plan (SRP) (NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," issued in 2007). This DSRS chapter reflects some important lessons the staff learned when using the SRP to review new large light-water reactor (LWR) designs.

The staff has incorporated the following lessons learned into this guidance:

1. This guidance emphasizes fundamental I&C design principles of independence, redundancy, predictability and repeatability, and diversity and defense-in-depth (D3). The staff intends to verify an applicant has shown the instrumentation and control (I&C) design incorporates these principles through analysis, such as hazard analysis. These principles are cornerstones of the staff's review in this area. The current SRP guidance is system-focused and does not take advantage of such a unifying framework. This guidance aims to address all the significant aspects of the I&C design in a unified manner through this framework.

2. This guidance highlights only those I&C requirements[1] and guidance applicable to the NuScale SMR. The existing SRP discusses regulatory requirements that are

[1] The design of digital I&C systems is governed by the legal requirements set forth in NRC regulations, including those in several of the General Design Criteria in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix A, and 10 CFR 50.55a(h), which incorporates by reference Institute of Electrical and Electronics Engineers (IEEE) Std 603-1991. NRC guidance endorses other IEEE standards, and these IEEE standards, as well as IEEE Std 603-1991, are written in terms of so-called system, functional, performance, design, and other "requirements." These terms are well-understood in the I&C technical community, but, except as used in IEEE Std 603-1991, are not legal requirements. To avoid confusion, this DSRS section will use the "requirements" terminology of the IEEE standards that are not incorporated into NRC regulations in connection with references to such standards. These "requirements," as referenced in this DSRS section, should be understood as recommendations that NRC staff considers adequate to satisfy portions of NRC regulatory requirements, but which are not the only acceptable methods of compliance.

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inapplicable to the NuScale SMR design and guidance that is not used in this DSRS. For example, the SRP cites the Institute of Electrical and Electronics Engineers (IEEE) Standard (Std) 279, "Criteria for Protection Systems for Nuclear Power Generating Stations," which is only applicable to nuclear power plants with construction permits issued after January 1, 1971, but before May 13, 1999.

3. The structure of this guidance reflects an integrated I&C design using digital technology, which is common in new and advanced reactor designs. In addition, the areas most significant to safety are discussed first. The current SRP guidance is system-based; therefore, many regulatory requirements and their supporting guidance are repeated in multiple subsections. The approach of this DSRS minimizes such repetition.

4. This guidance applies to microprocessor-based technology as well as other forms of complex logic such as programmable logic devices (e.g., field programmable gate arrays (FPGAs)). This DSRS uses the term software to refer to such technology and complex logic. The staff considers the information in this guidance sufficient to form a basis for an NRC finding in the area of software. The current SRP guidance is not always clear on the subject of software development because it reflects the complete software development life cycle, which may not be fully put into place at the design certification (DC) review stage.

5. This guidance introduces the use of an integrated hazards analysis approach, which is a well-established safety engineering practice, to the NRC's I&C reviews. This approach consolidates the various methods discussed in the current SRP and provides a consistent, comprehensive, and systematic way to address the potential hazards associated with the I&C systems in a unified framework.

6. This guidance also provides an approach to evaluate whether simplicity[2] has been considered in the design of the digital I&C system. Although no specific regulations, standards, or guidance explicitly address the concept of simplicity for digital I&C systems, recent experience in reviews of LWR applications has shown that complex I&C systems can challenge the demonstration of conformance with safety system design criteria such as independence. In this context, simplicity supports all fundamental design principles for developing I&C safety systems.

7. This guidance encompasses all relevant branch technical positions discussed in the current SRP. This guidance also clarifies the interface between the I&C area and other disciplines, such as equipment qualification (Chapter 3), human factors engineering (Chapter 18), quality assurance (Chapter 17), and reactor systems (Chapter 15).

<u>I&C System Review Scope</u>

The guidance contained in DSRS Chapter 7 covers all I&C safety systems and components (i.e., hardware, software, firmware, and other forms of complex logic). This guidance also

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The system, functional, performance, design, and other requirements of IEEE Std 603-1991, which are legal requirements, will be explicitly identified as originating from IEEE Std 603-1991.

[2] On October 14, 2008, in Volume 73 of the *Federal Register* (FR), pages 60612–60616 (73 FR 60612–60616), the Commission issued a policy statement on the regulation of advanced reactors [NRC-2008-0237].

covers those areas such as software tools and equipment that are used for the I&C design or are connected to the I&C systems or components for testing.

Most of the guidance in DSRS Chapter 7 is derived from IEEE Std 603-1991, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations," which is an NRC requirement incorporated by reference into Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.55a(h) for I&C safety systems. The scope of IEEE Std 603-1991 includes all I&C safety systems. Although IEEE Std 603-1991 does not establish requirements for I&C systems that are not safety-related, such as control systems and diverse I&C systems, the criteria in IEEE Std 603-1991 can be applied to any I&C system. Consequently, the reviewer will use the concepts of IEEE Std 603-1991 and the guidance in DSRS Chapter 7 as a starting point in the review of I&C systems that are not safety-related but are risk-significant, using a graded approach commensurate with the safety and risk significance of the system or component. Applicable review considerations include, for example, design bases, redundancy, independence, single failures, qualification, bypasses, status indication, and testing.

The guidance in Chapter 7 of the DSRS applies to microprocessor-based technology as well as other forms of complex logic such as programmable logic devices (e.g., FPGAs). Careful consideration should be given to characteristics of software elements (e.g., software/logic development process, effect of design errors, translation of algorithms, etc.) and hardware elements (e.g., failure modes, electronic-level timing, electrical issues, type of processing, etc.) for each type of technology chosen by the applicant.

<u>I&C System Review Objectives</u>

The objective of all I&C safety system reviews is to confirm (1) the I&C system design includes the functions necessary to assure adequate safety during operation of a nuclear power plant under normal conditions and under accident conditions, (2) these functions, and the I&C systems and equipment have been properly classified, and (3) an application demonstrates appropriate quality standards will be used for the design, fabrication, construction, and testing of I&C systems and equipment commensurate with the importance of the I&C safety functions to be performed.

To ensure the review objectives are met, the reviewer should confirm (1) variables and systems are properly monitored to assure a safe state, (2) variables and systems are maintained within their prescribed operating ranges, (3) variables and systems in an abnormal condition are identified and such an abnormal condition is communicated to the respective destinations credited in the safety analysis, (4) systems and components are automatically initiated to assure that fuel design limits are not exceeded because of anticipated operational occurrences (AOOs), and (5) systems are capable of operating under accident conditions.

DSRS Chapter 7 covers the following topics:

1. DSRS Section 7.1 offers guidance to I&C reviewers that is used to confirm the application contains sufficiently detailed design information, in the form of functional block diagrams, descriptions of operation, architectural descriptions, and other design details, to demonstrate the hardware and software for digital I&C architectures incorporate the fundamental design principles, namely independence; redundancy; predictability and repeatability; and D3.

2. DSRS Section 7.2 provides guidance associated with major functional and design characteristics, including IEEE Std 603-1991 performance requirements, general

arrangements, and materials of construction of I&C systems and components, all of which I&C reviewers will use to confirm the final design will conform to the design bases with adequate margin.

3. Sections 7.1 and 7.2, and Appendices A, B, and C of the DSRS are used in the review of an application to confirm all safety functions allocated to I&C safety-related systems, including the computer software supporting system operation, and all functions, information, and resources upon which these are dependent, are identified and analyzed in Chapter 7 in the application. The safety systems and functions supported by the I&C system are identified and described in other sections of the application (particularly in Chapters 5, 6, 8, 9, 10, 15, 17, 18, and 19). The review of these systems is coordinated (as described above) with the organizations that have primary review responsibility for the supported systems.

4. For DC and combined license (COL) reviews, the staff reviews the applicant's proposed inspections, tests, analyses, and acceptance criteria (ITAAC) associated with the structures, systems and components (SSCs) related to this DSRS section in accordance with SRP Section 14.3, "Inspections, Tests, Analyses, and Acceptance Criteria." The staff recognizes the review of ITAAC cannot be completed until after the rest of this portion of the application has been reviewed against the acceptance criteria in this DSRS 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 and DSRS Section 14.3.5.

DSRS Table 7-1 lists all regulatory requirements and applicable guidance associated with I&C safety systems, the applicable DSRS section, and I&C review responsibilities.

When an application takes exception to the guidelines applicable to I&C safety systems, the bases for such an exception is reviewed to confirm it is acceptable. The bases for each exception to the guidelines should demonstrate the exception does not result in a significant reduction in the margin of safety or in nonconformance with applicable requirements.

I&C System Review Interfaces

I&C systems provide for the collection, integration, and dissemination of information and the subsequent control actions needed to assure adequate safety during plant operation. These I&C functions involve numerous interfaces and interactions with other plant systems, necessitating corresponding interactions between the I&C discipline with other disciplines for review of a nuclear power plant design. Among these interfaces, the highest emphasis should be placed on the interface with Chapter 15, in which design-basis accidents and AOOs analyses are presented. These analyses establish the bases for safety system design and associated safety margins. For example, the Chapter 15 portion of the applicant's final safety analysis report (FSAR) identifies the variables to be monitored, the suitability of the monitored variables for generating signals to initiate automatic protective actions, and the credited automatic protective actions. The organization responsible for reviewing Chapter 15 of the application is the lead for completing this review, confirms all the safety functions credited from this perspective are adequately identified, and will request assistance from the I&C organization if needed.

Several other DSRS sections identify additional variables and control features with respect to a wide variety of SSCs. The organizations responsible for these reviews have the lead for completing them and will request assistance from the I&C organization if needed.

The reviews associated with Chapter 7 confirm the I&C system requirements, including those related to parameters and control features identified in other sections of the DSRS, are allocated to the protection and control systems. In some cases, I&C system components must meet specialized NRC regulatory requirements (such as environmental qualifications), so those components need to be reviewed by other organizations. These other organizations have the lead for completing the special requirement reviews and will request assistance from the I&C organization if needed.

The following organizations provide the lead role in evaluating the interfaces and interactions described. I&C reviewers support these reviews when requested by the lead organization. Specific technical questions on safety or compliance with NRC regulatory requirements may warrant additional interactions between organizations to resolve the concerns.

1. The organization responsible for the review of transient and accident analyses evaluates the adequacy of limiting conditions for operation, limiting safety system settings, and design descriptions for safety-related components and systems. The I&C reviewer ensures that the application lists the settings of all the protection and safety system functions that are credited in the safety analysis and that the variables monitored to support these functions are appropriate (Chapter 15).

2. The organization responsible for the review of reactor systems evaluates the adequacy of protective, control, display, and interlock functions and confirms they are consistent with the accident analysis, the operation of the I&C systems, and the requirements of General Design Criteria (GDCs) 10, 15, 28, 33, 34, and 35 (Chapter 5).

3. The organization responsible for the review of plant systems evaluates the adequacy of the auxiliary supporting features and other auxiliary features to assure that they satisfy the applicable acceptance criteria. These features include, for example, compressed (instrument) air, cooling water, systems for boration of reactor or spent fuel pool makeup water, lighting, heating, and air conditioning. This review confirms (1) the design of the auxiliary supporting features and other auxiliary features ensure that these components, equipment, and systems do not degrade the I&C safety systems below an acceptable level, and (2) the auxiliary supporting features and other auxiliary features will maintain the environmental conditions in the areas containing I&C equipment as specified in the FSAR. This review includes the design criteria and testing methods employed in the seismic design and installation of equipment implementing auxiliary supporting features and other auxiliary features. The organization responsible for review of plant systems also evaluates the adequacy of protective, control, display, and interlock functions, and confirms they are consistent with the operation of the supported system credited in the safety analysis and the requirements of GDCs 41 and 44 (Chapter 9).

4. The organization responsible for the review of containment systems reviews the containment ventilation and atmospheric control systems provided to maintain environmental conditions for I&C equipment located inside containment. This organization also evaluates the adequacy of protective, control, display, and interlock functions associated with containment systems and severe accidents, and confirms they are consistent with the accident analysis, operation of containment features, and the requirements of GDCs 16 and 38 (Chapter 6).

5. The organization responsible for the review of electrical systems (1) evaluates the adequacy of physical separation criteria for cabling and electrical power equipment,

(2) determines whether power supplied to redundant systems is supplied by appropriately redundant sources, and (3) confirms the adequacy of design features associated with the proper functioning of the onsite and offsite power systems, such as protective devices. The guidance of DSRS Chapter 7 also applies to any protective device, such as a circuit breaker or relay with digital logic built into it. The guidance of DSRS Chapter 7 also applies to any grounding paths from an I&C element in a safety system to a ground through the electrical power network (Chapter 8).

6. The organization responsible for the review of environmental qualification reviews the environmental qualification of I&C equipment. The scope of this review includes the design criteria and qualification testing methods and procedures for I&C equipment consistent with GDC 4, 10 CFR 50.49 and Section 5.4 of IEEE Std 603-1991 (Chapter 3).

7. The organization responsible for the review of seismic qualification reviews the seismic qualification demonstration for I&C equipment, including the design criteria and qualification testing methods and procedures consistent with 10 CFR Part 50, Appendix B, Criterion III (Chapter 3).

8. The organization responsible for the review of human-machine interface evaluates the adequacy of the arrangement and location of I&C, and confirms the functions allocated to the operators can be successfully accomplished (Chapter 18).

9. The organization responsible for the review of quality assurance reviews general quality assurance programs (Chapter 17).

10. The organization responsible for the review of probabilistic risk analysis and severe accidents evaluates the adequacy of the models and methods used for the probabilistic risk analysis and strategies for handling severe accidents, including aspects associated with I&C (Chapter 19).

<u>DSRS Chapter 7 Acceptance Criteria and Review Process</u>

1. Regulatory Requirements

The regulations in 10 CFR 50.55a(h) require compliance with IEEE Std 603-1991, including the correction sheet dated January 30, 1995, which is referenced in 10 CFR 50.55a(h)(2) and (3). The standard sets forth design and functional requirements that are discussed in this DSRS. In addition, IEEE Std 7-4.3.2, "IEEE Standard for Digital Computers in Safety Systems of Nuclear Power Generating Stations," provides specific guidance for the application of IEEE Std 603-1991 criteria to computer-based I&C systems as endorsed by the version of Regulatory Guide (RG) 1.152, "Criteria for Digital Computers in Safety Systems of Nuclear Power Plants," in place 6 months before the docket date of the application.

In accordance with 10 CFR 50.55a(a)(3), an applicant can propose alternatives to the requirements of 10 CFR 50.55a(h), but the applicant must demonstrate the proposed alternative would provide an acceptable level of quality and safety or that compliance with the specified requirements of 10 CFR 50.55a(h) would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. In accordance with 10 CFR 52.47(a)(8), (21), and (22), for new reactor license applications submitted under Part 52, the applicant is required to include the following information:

(1) the proposed technical resolution of unresolved safety issues (USIs) and medium- and high-priority generic safety issues (GSIs) that are identified in the version of NUREG-0933, "Resolution of Generic Safety Issues (Formerly titled 'A Prioritization of Generic Safety Issues')," current on the date 6 months before the docket date of the application and that are technically relevant to the design, (2) the information necessary to demonstrate how operating experience insights have been incorporated into the plant design, and (3) the information necessary to demonstrate compliance with any technically relevant portions of the Three Mile Island (TMI) requirements set forth in 10 CFR 50.34(f), except paragraphs (f)(1)(xii), (f)(2)(ix), and (f)(3)(v). These cross-cutting review areas should be addressed by the reviewer for each technical subsection and relevant conclusions documented in the corresponding safety evaluation report section.

2. DSRS Acceptance Criteria

Specific DSRS acceptance criteria acceptable to meet the relevant requirements of the RC's regulations identified above are set forth below. The DSRS is not a substitute for the NRC's regulations, and compliance with it is not required. As an alternative, and as described in more detail below, an applicant may identify the differences between a DSRS section and the design features (DC and COL applications only), analytical techniques, and procedural measures proposed in an application and discuss how the proposed alternative provides an acceptable method of complying with the NRC regulations that underlie the DSRS acceptance criteria.

3. Level of Review Applied to I&C Systems

As stated in Commission Paper SECY-11-0024, "Use of Risk Insights to Enhance the Safety Focus of Small Modular Reactor Reviews," the level of review for a particular SSC is derived from both the SSC's safety importance (i.e., safety-related or nonsafety-related) and risk significance. The introduction to NUREG-0800, "Introduction – Part 2: Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: Light-Water Small Modular Reactor Edition" (NUREG-0800, Intro Part 2) describes the licensing review philosophy and framework to be applied by the staff for new integral pressurized water reactor iPWR DC and COL applications under 10 CFR Part 52. With the incorporation of risk insights, I&C systems may be classified as:

- safety-related risk-significant (A1)
- safety-related nonrisk-significant (A2)
- nonsafety-related risk-significant (B1)
- nonsafety-related nonrisk-significant (B2)

The staff expects the NuScale SMR application will include the classification of SSCs, a list of risk-significant SSCs, and a list of SSCs subject to Regulatory Treatment of Nonsafety Systems (RTNSS) (called RTNSS SSCs). The I&C staff will support a review of RTNSS SSCs with other technical organizations in accordance with the guidance in SRP Sections 3.2, 17.4, and 19.3 to confirm nonsafety-related SSCs with risk-significant functions are included within the scope of the RTNSS process.

In the context of I&C, the term "safety system" is used to include all systems that are safety related. Protection systems are I&C safety systems that initiate actions to assure that fuel design limits are not exceeded because of AOOs and respond to design basis

events (DBEs). During safe shutdown,[3] reactivity control systems must be capable of maintaining the core in a subcritical condition under cold conditions, and residual heat removal systems must be capable of maintaining adequate cooling of the core.

i. The reactor trip system (RTS) initiates rapid control rod insertion to mitigate the consequences of AOOs and DBEs.

ii. The engineered safety features actuation system (ESFAS) initiates and controls safety equipment that removes heat or otherwise assists with maintaining the integrity of the physical barriers to radioactive release (e.g., fuel cladding, reactor coolant pressure boundary, and containment). Typical engineered safety features (ESF) systems include:

   a. containment and reactor vessel isolation systems
   b. emergency core cooling systems
   c. containment heat removal and depressurization systems
   d. pressurized-water reactor (PWR) auxiliary feedwater systems
   e. emergency boration systems
   f. containment air purification and cleanup systems
   g. containment combustible gas control systems
   h. control room isolation and emergency heating, ventilating, and air conditioning

iii. Safe shutdown systems function to achieve and maintain a safe shutdown condition of the plant. The safe shutdown systems include I&C systems used to maintain the reactor core in a subcritical condition and provide adequate core cooling to achieve and maintain both hot and cold shutdown conditions, as defined in SECY 95-132 “Policy and Technical Issues Associated with the Regulatory Treatment of Nonsafety Systems in Passive Plant Designs (SECY 94-084).” Typical safe shutdown functions include:

   a. reactivity control
   b. reactor coolant makeup
   c. reactor pressure control
   d. decay heat removal

   To the extent ESF systems are used to achieve and maintain safe shutdown, the review of these systems is limited to those features that are unique to safe shutdown and not credited for accident mitigation.

iv. Auxiliary supporting features and other auxiliary features are systems or components of systems that provide support functions necessary for the safety systems to accomplish their safety functions. Figure 3 of IEEE Std 603-1991, “Examples of Equipment Fitted to Safety System Scope Diagram,” provides a matrix with an extensive list of auxiliary

[3] The NRC considers a “safe stable shutdown condition” for advanced passive LWRs to be a condition by which all plant conditions are stable and within regulatory limits and the reactor coolant system pressure is stabilized and reactor coolant temperature is less than or equal to 215 degrees Celsius (C) (420 degrees Fahrenheit (F)).

supporting features and other auxiliary features. Heating, ventilation, and air conditioning systems and electrical power systems are examples of auxiliary supporting features. Auxiliary supporting features are discussed primarily in Chapters 8 and 9 of the Safety Analysis Report (SAR). Examples of other auxiliary features include built-in test equipment and isolation devices. The I&C aspects of auxiliary supporting features and other auxiliary features are addressed in the review of those SAR sections that discuss the systems or components that provide these functions. To the extent the operation of auxiliary supporting features or other auxiliary features are initiated by the protection system, this aspect is included in the review of I&C safety systems.

With this determination, the review framework for I&C systems will be carried out as follows:

A. For SSCs determined to be safety-related risk-significant (A1), the level of review will involve detailed analyses and evaluation techniques to satisfy the acceptance criteria specified in the DSRS. This includes Sections 7.1 and 7.2, and Appendices A, B, and C of Chapter 7 of the DSRS.

B. For SSCs determined to be safety-related nonrisk-significant (A2), the staff should use NUREG-0800, Intro – Part 2 to inform the identification and treatment of safety-related nonrisk-significant SSCs. Reviewers should discuss this categorization and review approach with management in their technical areas. The review will identify those requirements that may apply to an SSC and that could be used to demonstrate the satisfaction of design-based or performance-based acceptance criteria. The staff should consider using selected requirements to determine if applicable design-based and performance-based acceptance criteria have been satisfied. Any remaining criteria that are not satisfied will involve detailed analyses and evaluation techniques to satisfy the acceptance criteria. This would include appropriate sections from Sections 7.1 and 7.2, and Appendices A, B, and C of Chapter 7 of the DSRS.

C. For SSCs determined to be nonsafety-related risk-significant (B1), the level of review will shift from applying analyses and evaluation techniques to identifying those programmatic elements applicable to I&C systems that satisfy the acceptance criteria in the DSRS. The objectives of the review are to confirm B1 systems are capable of controlling variables within prescribed operating ranges, and to confirm the effects of operation or failures of these systems are bounded by the accident analyses in Chapter 15 of the DSRS.

   Staff expects RTNSS systems to be in the scope of the B1 systems. Not all B1 systems are RTNSS, but the B1 acceptance criteria outlined below will be used for systems and functions that are considered risk-significant. The RTNSS criteria used to determine risk-significant SSC functions are contained in Section 19.3 of the SRP. The I&C technical staff assist in the review of those SSC functions associated with the following RTNSS categories:

   i. RTNSS “A”—SSC functions relied on to meet beyond design basis deterministic performance requirements such as those set forth in 10 CFR 50.62 for mitigating Anticipated Transients Without Scram (ATWS) and in 10 CFR 50.63 for Station Blackout. The I&C review scope includes

the diverse actuation system that is used to actuate plant systems for ATWS mitigation.

ii. RTNSS "B"—SSC functions relied on to ensure long-term safety (beyond 72 hours) and to address seismic events. The I&C review scope includes post-accident monitoring systems, including safety-related displays in the control room, emergency lighting, control room cooling to remove heat generated by personnel, and monitoring equipment.

iii. RTNSS "C"—SSC functions relied on under power-operating and shutdown conditions to meet the Commission's safety goal guidelines of a core damage frequency of less than 1×10^{-4} each reactor year and a large release frequency of less than 1×10^{-6} each reactor year.

iv. RTNSS "D"—SSC functions needed to meet the containment performance goal, including containment bypass, during severe accidents.

v. RTNSS "E"—SSC functions relied on to prevent significant adverse systems interactions between passive safety systems and active nonsafety SSCs. The I&C review scope includes evaluations of the potential for adverse interaction between passive safety-related and active nonsafety-related systems to confirm any nonsafety-related design features or functional capabilities relied upon to prevent nonsafety-related systems from impairing a safety function have been included in the scope of RTNSS.

There may be other nonsafety-related SSCs whose functions could affect plant safety and control that are not considered within the scope of RTNSS. Examples include systems used for reactivity control of the reactor through the positioning of the control rods, systems used to control the feedwater to the reactor vessel and feedwater temperature, and systems used to regulate reactor steam flow and pressure. These systems can affect the performance of safety-related functions either through normal operation, inadvertent operation, or various AOOs that could be considered candidates for regulatory oversight. If such systems and functions are considered risk-significant, the I&C staff will conduct a review using the review criteria for B1 SSCs.

The I&C review of B1 SSCs will emphasize the following specific topics from Section 19.3 of the SRP and selected topics from Sections 7.1 and 7.2, and Appendices A, B, and C of Chapter 7 of the DSRS:

i. The reviewer should help with the identification of SSC functions based on the RTNSS criteria listed above.

ii. The reviewer should review the functional design of RTNSS SSCs, including the adequacy of functional design and design improvements to minimize adverse interaction between safety passive and nonsafety-related active systems. The reviewer will confirm the following:

a. The reviewer should confirm the nonsafety-related systems meet the reliability and availability goals assumed for the system and

that a single point of failure (i.e., a software error or a failure of a single component in a nonsafety system would not result in consequences more severe than those described in the analysis in Chapter 15 of the SAR.

b. The reviewer should review the bases for the nonsafety-related systems' design to confirm the necessary features for manual and automatic control of process variables are within prescribed normal operating limits.

c. The reviewer should confirm the plant accident analysis in Chapter 15 of the SAR does not rely on the operability of any nonsafety-related system function to ensure that regulatory limits are met.

d. For nonsafety-related system elements credited in the D3 analysis, the reviewer should use the review criteria for diverse I&C systems in DSRS Section 7.1.5.

e. The reviewer should confirm the safety analysis includes consideration of the effects of both nonsafety-related system action and inaction in assessing the transient response of the plant for postulated accidents and AOOs.

f. The reviewer should confirm the failure of any nonsafety-related system component or any auxiliary supporting system for nonsafety-related systems does not cause plant conditions more severe than those described in the analysis of AOOs in Chapter 15 of the application. This evaluation should address failure modes that can be associated with digital systems such as software design errors as well as random hardware failures (the reviewer need not evaluate multiple independent failures).

g. The reviewer should confirm the consequential effects of AOOs and postulated accidents do not lead to nonsafety-related system failures that would result in consequences more severe than those described in the analysis in Chapter 15 of the SAR.

h. The reviewer should confirm I&C systems include environmental control as necessary to protect equipment from environmental extremes. This would include, for example, heat tracing of instruments and instrument sensing lines as discussed in RG 1.151, "Instrument Sensing Lines," and cabinet cooling fans.

i. With respect to an I&C system that is nonsafety-related, the reviewer will confirm the application describes quality measures commensurate with the importance of the system function to be accomplished. Refer to DSRS Section 7.2.1 for additional guidance. To satisfy GDC 1, an applicant may choose to apply its Appendix B Quality Assurance (QA) program to I&C systems that are nonsafety-related. In any case, the development of a software-based I&C system that is nonsafety-related should follow

a structured system and software development framework consistent with the guidance in this section.

j. The reviewer should use the review criteria for independence in DSRS Section 7.1.2 to confirm adequate independence of safety systems from nonsafety-related systems.

k. The nonsafety-related systems design should minimize the potential for inadvertent actuation and challenges to safety-related systems.

l. The reviewer should use the review criteria for access control in DSRS Section 7.2.9 to confirm adequate physical and electronic control of access to digital computer-based nonsafety-related system software and data to prevent changes by unauthorized personnel. Control should address access though network connections and through maintenance equipment.

iii. The reviewer should review the proposed regulatory treatment proposed for SSCs in the scope of the RTNSS program to confirm the oversight is commensurate with the risk significance of each SSC's reliability/availability mission.

Note that for SSCs determined to be highly risk-significant, a more detailed review may be appropriate using Sections 7.1 and 7.2, and Appendices A, B, and C of Chapter 7 of the DSRS.

D. For SSCs determined to be nonsafety-related nonrisk-significant (B2), both the design-related review and the programmatic elements are anticipated to be minimal. For the performance-oriented acceptance criteria, the review is focused on identifying those performance-based activities (e.g., tests or inspections) within the applicable programmatic requirements that can be used to satisfy the acceptance criteria from the DSRS.

TABLE 7.1 INSTRUMENTATION AND CONTROLS—MAPPING OF REGULATORY REQUIREMENTS, GUIDANCE AND DSRS REVIEW CRITERIA

Regulations / Guidance	Location in SRP/DSRS, as applicable for this design	Review Responsibilities
10 CFR 50.55a(h)		
IEEE Std 603-1991, Section 4, "Safety System Designation"	7.1.1 Safety System Design Basis 7.1.4 Predictability and Repeatability (covers Section 4.10)	Full
IEEE Std 603-1991, Section 5.1, "Single-Failure Criterion"	7.1.3 Redundancy 7.1.5 Diversity and Defense-in-Depth	Full
IEEE Std 603-1991, Section 5.2, "Completion of Protective Action"	7.2.3 Reliability, Integrity, and Completion of Protective Action	Full
IEEE Std 603-1991, Section 5.3, "Quality"	Covered in Chapter 17 of the SRP	Partial, [1]
IEEE Std 603-1991, Section 5.4, "Equipment Qualification"	7.2.2 Equipment Qualification	Partial, [2]

Regulations / Guidance	Location in SRP/DSRS, as applicable for this design	Review Responsibilities
IEEE Std 603-1991, Section 5.5, “System Integrity”	7.2.3 Reliability, Integrity, and Completion of Protective Action	Full
IEEE Std 603-1991, Section 5.6, “Independence”	7.1.2 Independence	Full
IEEE Std 603-1991, Section 5.7, “Capability for Test and Calibration”	7.2.15 Capability for Test and Calibration	Full
IEEE Std 603-1991, Section 5.8, “Information Displays”	7.2.4 Operating and Maintenance Bypasses 7.2.13 Displays and Monitoring	Full
IEEE Std 603-1991, Section 5.9, “Control of Access”	7.2.9 Control of Access, Identification, and Repair	Full
IEEE Std 603-1991, Section 5.10, “Repair”	7.2.9 Control of Access, Identification, and Repair	Full
IEEE Std 603-1991, Section 5.11, “Identification”	7.2.9 Control of Access, Identification, and Repair	Full

Regulations / Guidance	Location in SRP/DSRS, as applicable for this design	Review Responsibilities
IEEE Std 603-1991, Section 5.12, “Auxiliary Features”	7.2.8 Auxiliary Features	Full
IEEE Std 603-1991, Section 5.13, “Multi-Unit Stations”	7.2.11 Multi-Unit Stations	Full
IEEE Std 603-1991, Section 5.14, “Human Factors Considerations”	7.2.14 Human Factors Considerations	Full
IEEE Std 603-1991, Section 5.15, “Reliability”	7.2.3 Reliability, Integrity, and Completion of Protective Action	Full
IEEE Std 603-1991, Section 6.1, “Automatic Control”	7.2.12 Automatic and Manual Control	Full
IEEE Std 603-1991, Section 6.2, “Manual Control”	7.2.12 Automatic and Manual Control	Full
IEEE Std 603-1991, Section 6.3, “Interaction Between the Sense and Command Features and Other Systems”	7.2.10 Interaction between Sense and Command Features and Other Systems	Full

Regulations / Guidance	Location in SRP/DSRS, as applicable for this design	Review Responsibilities
IEEE Std 603-1991, Section 6.4, “Derivation of System Inputs”	7.2.6 Derivation of System Inputs	Full
IEEE Std 603-1991, Section 6.5, “Capability for Testing and Calibration”	7.2.15 Capability for Test and Calibration	Full
IEEE Std 603-1991, Section 6.6, “Operating Bypasses”	7.2.4 Operating and Maintenance Bypasses	Full
IEEE Std 603-1991, Section 6.7, “Maintenance Bypass”	7.2.4 Operating and Maintenance Bypasses	Full
IEEE Std 603-1991, Section 6.8, “Setpoints”	7.2.7 Setpoints	Full
IEEE Std 603-1991, Section 7.1, “Automatic Control”	7.2.12 Automatic and Manual Control	Full
IEEE Std 603-1991, Section 7.2, “Manual Control”	7.2.12 Automatic and Manual Control	Full
IEEE Std 603-1991, Section 7.3, “Completion of Protective Action”	7.2.3 Reliability, Integrity, and Completion of Protective Action	Full

Regulations / Guidance	Location in SRP/DSRS, as applicable for this design	Review Responsibilities
IEEE Std 603-1991, Section 7.4, “Operating Bypass”	7.2.4 Operating and Maintenance Bypasses	Full
IEEE Std 603-1991, Section 7.5, “Maintenance Bypass”	7.2.4 Operating and Maintenance Bypasses	Full
10 CFR Part 50, Appendix A, GDC		
GDC 1, “Quality standards and records”	Covered in Chapter 17 of the DSRS	Partial, [1]
GDC 2, “Design bases for protection against natural phenomena”	7.2.2 Equipment Qualification Coordinated with Chapter 3 of the SRP and DSRS	Partial, [3]
GDC 4, “Environmental and dynamic effects design bases”	7.2.2 Equipment Qualification Coordinated with Chapter 3 of the SRP and DSRS	Partial, [4]
GDC 10, “Reactor design”	7.1.1 Safety System Design Basis Coordinated with Chapter 4 of the SRP and DSRS	Partial, [5]

Regulations / Guidance	Location in SRP/DSRS, as applicable for this design	Review Responsibilities
GDC 13, “Instrumentation and control”	Sections 7.1 and 7.2 of the DSRS	Full, [6]
GDC 15, “Reactor coolant system design”	7.1.1 Safety System Design Basis Coordinated with Chapter 5 of the SRP and DSRS	Partial, [7]
GDC 16, “Containment design”	7.1.1 Safety System Design Basis Coordinated with Chapter 6 of the SRP and DSRS	Partial, [8]
GDC 19, “Control room”	Sections 7.1 and 7.2 of the DSRS Coordinated with Chapter 6 of the SRP and DSRS Coordinated with Chapter 18 of the SRP	Full, [9], [25], [26]
GDC 20, “Protection system functions”	Sections 7.1 and 7.2 of the DSRS	Full, [10]
GDC 21, “Protection system reliability and testability”	7.1.2 Independence 7.1.3 Redundancy 7.1.4 Predictability and Repeatability 7.2.15 Capability for Test and Calibration	Full, [11]

Regulations / Guidance	Location in SRP/DSRS, as applicable for this design	Review Responsibilities
GDC 22, “Protection System Independence”	7.1.2 Independence 7.1.5 Diversity and Defense-in-Depth	Full, [12]
GDC 23, “Protection system failure modes”	7.1.1 Safety System Design Basis Appendix A, Hazard Analysis	Full, [13]
GDC 24, “Separation of Protection and Control Systems”	7.1.2 Independence 7.1.3 Redundancy 7.1.5 Diversity and Defense-in-Depth	Full, [14]
GDC 25, “Protection system requirements for reactivity control malfunctions”	Coordinated with Chapter 4 of the SRP and DSRS	Partial, [15]
GDC 28, “Reactivity limits”	Coordinated with Chapter 15 of the SRP and DSRS	Partial, [16]
GDC 29, “Protection against Anticipated Operational Occurrences”	7.1.4 Predictability and Repeatability	Full, [17]
GDC 64, “Monitoring Radioactivity Releases ”	7.2.13 Displays and Monitoring	Full

Regulations / Guidance	Location in SRP/DSRS, as applicable for this design	Review Responsibilities
10 CFR 50.34(f)(2), which Addresses TMI Requirements		
10 CFR 50.34(f)(2)(iv) (Safety Parameter Display Console)	7.1.5 Diversity and Defense-in-Depth 7.2.13 Displays and Monitoring	Full
10 CFR 50.34(f)(2)(v) (Bypass and Inoperable Status Indication)	7.2.4 Operating and Maintenance Bypasses 7.2.13 Displays and Monitoring	Full, [18]
10 CFR 50.34(f)(2)(xi) (Direct Indication of Relief and Safety Valve Position)	7.2.13 Displays and Monitoring	Full, [19]
10 CFR 50.34(f)(2)(xii) (Auxiliary Feedwater System Automatic Initiation and Flow Indication)	7.2.13 Displays and Monitoring	Full, [20]
10 CFR 50.34(f)(2)(xvii) (Accident Monitoring Instrumentation)	7.2.13 Displays and Monitoring	Full

Regulations / Guidance	Location in SRP/DSRS, as applicable for this design	Review Responsibilities
10 CFR 50.34(f)(2)(xviii) (Instrumentation for the Detection of Inadequate Core Cooling)	7.2.13 Displays and Monitoring	Full, [19]
10 CFR 50.34(f)(2)(xiv) (Containment Isolation Systems)	7.1.5 Diversity and Defense-in-Depth Paragraphs (B) and (D) of 50.34(f)(2)(xiv) should be coordinated with Chapter 6 of the SRP and DSRS	Partial, [21]
10 CFR 50.34(f)(2)(xix) (Instruments for Monitoring Plant Conditions Following Core Damage)	7.2.13 Displays and Monitoring	Full
10 CFR 50.34(f)(2)(xx) (Power for Pressurizer Level Indication and Controls for Pressurizer Relief and Block Valves)	7.2.13 Displays and Monitoring Coordinated with Chapter 8 of the SRP and DSRS	Partial, [22]
10 CFR 50.34(f)(2)(xxii) (Failure Mode and Effect Analysis of Integrated Control System)	7.2.15 Capability for Test and Calibration	Full

Regulations / Guidance	Location in SRP/DSRS, as applicable for this design	Review Responsibilities
10 CFR 50.34(f)(2) (xxiii) (Anticipatory Trip on Loss of Main Feedwater or Turbine Trip)	7.2.8 Auxiliary Features	Full
Other Regulations		
10 CFR 50.55a(a)(1), "Quality Standards for Systems Important to Safety"	Covered in Chapter 17 of the SRP	Partial, [1]
10 CFR 50.62, "Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants"	7.1.5 Diversity and Defense-in-Depth	Full, [23]
10 CFR 50.36(c)(1)(ii)(A) (Technical Specifications, Safety Limits, Limiting Safety System Settings, and Limiting Control Settings)	7.2.7 Setpoints	Full
10 CFR 50.36(c)(3), "Surveillance Requirements"	7.2.7 Setpoints 7.2.15 Capability for Test and Calibration	Full

Regulations / Guidance	Location in SRP/DSRS, as applicable for this design	Review Responsibilities
10 CFR 50.34(b)(2)(i) (Contents of Applications; Technical Information, Final Safety Analysis Report)	7.2.8 Auxiliary Features	Full
10 CFR 52.47(b)(1) (ITAAC)	DSRS Section 14.3.5	Full [27]
10 CFR 52.80(a) (COL ITAAC)	DSRS Section 14.3.5	Full [27]
10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants"	7.2.2 Equipment Qualification Coordinated with Chapter 3 of the SRP and DSRS	Partial, [24]
Regulatory Guides (Guidance)		
RG 1.22, "Periodic Testing of Protection System Actuation Functions"	7.2.15 Capability for Test and Calibration	

Regulations / Guidance	Location in SRP/DSRS, as applicable for this design	Review Responsibilities
RG 1.47, “Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems”	7.2.4 Operating and Maintenance Bypasses 7.2.13 Displays and Monitoring	
RG 1.53, “Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems”	7.1.3 Redundancy 7.1.5 Diversity and Defense-in-Depth 7.2.11 Multi-Unit Stations	
RG 1.62, “Manual Initiation of Protection Action”	7.1.5 Diversity and Defense-in-Depth 7.2.12 Automatic and Manual Control	
RG 1.75, “Criteria for Independence of Electrical Safety Systems”	7.1.2 Independence 7.2.9 Control of Access, Identification, and Repair	
RG 1.97, “Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants”	7.2.13 Displays and Monitoring	
RG 1.105, “Setpoints for Safety-Related Instrumentation”	7.2.7 Setpoints	

Regulations / Guidance	Location in SRP/DSRS, as applicable for this design	Review Responsibilities
RG 1.118, “Periodic Testing of Electric Power and Protection Systems”	7.2.15 Capability for Test and Calibration	
RG 1.151, “Instrument Sensing Lines”	7.2.2 Equipment Qualification	[2]
RG 1.152, “Criteria for Use of Computers in Safety Systems of Nuclear Power Plants”	7.1.2 Independence 7.2.2 Equipment Qualification [2] 7.2.3 Reliability, Integrity, and Completion of Protective Action 7.2.5 Interlocks 7.2.9 Control of Access, Identification, and Repair 7.2.11 Multi-Unit Stations	
RG 1.168, “Verification, Validation, Reviews, and Audits for Digital Computer Software Used in Safety Systems of Nuclear Power Plants”	7.2.1 Quality	
RG 1.169, “Configuration Management Plans for Digital Computer Software Used in Safety Systems of Nuclear Power Plants”	7.2.1 Quality	

Regulations / Guidance	Location in SRP/DSRS, as applicable for this design	Review Responsibilities
RG 1.170, “Software Test Documentation for Digital Computer Software Used in Safety Systems of Nuclear Power Plants”	7.2.1 Quality	
RG 1.171, “Software Unit Testing for Digital Computer Software Used in Safety Systems of Nuclear Power Plants”	7.2.1 Quality	
RG 1.172, “Software Requirements Specifications for Digital Computer Software Used in Safety Systems of Nuclear Power Plants”	7.2.1 Quality	
RG 1.173, “Developing Software Life Cycle Processes for Digital Computer Software Used in Safety Systems of Nuclear Power Plants”	7.2.1 Quality	
RG 1.180, “Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems”	7.2.2 Equipment Qualification	[2]

Regulations / Guidance	Location in SRP/DSRS, as applicable for this design	Review Responsibilities
RG 1.204, “Guidelines for Lightning Protection of Nuclear Power Plants”	7.2.2 Equipment Qualification	[2]
RG 1.209, “Guidelines for Environmental Qualification of Safety-Related Computer-Based Instrumentation and Control Systems in Nuclear Power Plants”	7.2.2 Equipment Qualification	[2]

Notes:

[1] This regulation is applicable to all I&C systems and components important to safety. The reviewer should confirm Chapter 17 identifies I&C safety systems and components that are subject to the QA requirements established in 10 CFR Part 50, Appendix B, 10 CFR 50.55a(a)(1), and GDC 1.

[2] The I&C review of equipment qualification is limited to a confirmation that I&C equipment (including isolation devices) subject to qualification requirements have been selected and identified in the application. Organizations responsible for seismic and environmental qualifications verify the functional performance requirements described in DSRS Chapter 3 are met.

[3] This regulation is applicable to all I&C safety systems and supporting data communication systems. The I&C review for GDC 2 should confirm the I&C systems important to safety are designed for protection against natural phenomena consistent with the analysis of these events as provided in Chapter 3 of the application, and that they are located and housed in structures consistent with these requirements. DSRS Section 7.2.2 addresses seismic qualification of I&C equipment, which is required by 10 CFR Part 50, Appendix B, Criterion III, 10 CFR 50.49, and Section 5.4 of IEEE Std 603-1991.

[4] This regulation is applicable to all I&C safety systems and supporting data communication systems. The design bases should identify those systems and components that are designed to accommodate the effects of environmental conditions and that are protected from the dynamic effects of missiles, pipe whipping, and discharging fluids. If systems or components are qualified to survive the environmental effects of postulated accidents for limited periods of time, the bases for limited operability should be provided. In a standard design proposed for certification, the I&C systems needed for severe accidents must be designed so there is reasonable assurance they will operate in the severe accident environment for which they are intended and over the time span for which they are needed. The review of this requirement should be coordinated with the organization responsible for review of environmental qualification. DSRS Section 7.2.2 addresses environmental qualification of I&C equipment, which is required by 10 CFR 50.49 and Section 5.4 of IEEE Std 603-1991.

[5] This regulation is applicable to I&C protection and control systems. The I&C review scope addresses the adequacy of I&C protective and control functions to confirm I&C systems are designed with sufficient margin to assure that specified fuel design limits are not exceeded.

[6] This regulation is applicable to all I&C systems including supporting data communication systems. The review of GDC 13 should determine the adequacy of the information provided for the RTS, ESFAS, ESF, safe shutdown, interlock, control, and diverse I&C systems over the anticipated ranges for normal operation, AOOs, and accident conditions.

[7] This regulation is applicable to I&C protection and control systems. The I&C review scope addresses the adequacy of I&C protective and control functions to confirm I&C systems are designed with sufficient margin to assure that the design conditions of the

reactor coolant pressure boundary are not exceeded. Evaluation of I&C system contributions to design margin for reactor coolant systems should be a part of the review of the adequacy of I&C protective and control functions.

[8] This regulation is applicable to ESF I&C systems. The review of GDC 16 should confirm the I&C systems provide the functions, performance, and reliability necessary to support the containment system safety function. GDC 16 imposes functional requirements on ESF I&C systems to the extent they support the requirement that the containment provide a leak tight barrier.

[9] This regulation is applicable to all I&C systems and supporting data communication systems.

[10] This regulation is applicable to I&C protection systems, RTS, and ESFAS.

[11] This regulation is applicable to I&C protection systems, RTS, ESFAS, and supporting data communication systems. Review of compliance with GDC 21 should address:

- design basis
- single-failure criterion
- completion of protective action
- quality
- system integrity
- physical, electrical, and communications independence
- capability for test and calibration
- indication of bypass
- control of access to safety system equipment
- repair and troubleshooting provisions
- identification of protection system equipment
- auxiliary features
- multiunit stations
- human factors considerations
- reliability
- manual controls
- derivation of system inputs
- operating bypasses
- maintenance bypasses
- setpoints

[12] This regulation is applicable to I&C protection systems, RTS, ESFAS, and supporting data communication systems. Review of compliance with GDC 22 should address:

- design basis reliability
- single-failure criterion
- quality
- equipment qualification
- system integrity
- physical, electrical, and communications independence
- manual controls
- setpoints

[13] This regulation is applicable to I&C protection systems, RTS, ESFAS, and supporting data communication systems.

[14] This regulation is applicable to all I&C systems.

[15] This regulation is applicable to the RTS and reactivity control system interlocks. For the review of GDC 25, the staff should confirm the protection system is designed for an appropriate spectrum of reactivity control system malfunctions as addressed in the review of protection system design basis. Chapter 15 of the application addresses the capability of the protection system to ensure that fuel design limits are not exceeded for events caused from malfunctions of the reactivity control systems.

[16] This regulation is applicable to I&C interlock and control systems. The review of GDC 28 should confirm the I&C systems provide the functions, performance, and reliability necessary to limit reactivity increases.

[17] This regulation is applicable to the protection systems, reactivity control functions of control systems, and supporting data communication systems. Probabilistic reliability assessments may be performed by NRC staff to provide a basis for development of deterministic criteria for specific systems.

[18] For compliance with 10 CFR 50.34(f)(2)(v), the staff should address the characteristics of IEEE Std 603-1991, Sections 5.6, 5.8, 5.12, and 6.3 for the safety system. If the staff review of the safety system shows it satisfies the criteria stated in DSRS Sections 7.1 and 7.2, then it also meets the requirements of 10 CFR 50.34(f)(2)(v). In addition, providing automatic indication of the bypassed and operable status of safety systems is covered as part of the staff's review of DSRS Section 7.2.13.

[19] NUREG-0737 provides additional guidance on conformance with this requirement.

[20] For compliance with 10 CFR 50.34(f)(2)(xii), the staff will, in addition to the review of DSRS Section 7.2.13, verify automatic and manual auxiliary feedwater (AFW) system initiation has been provided and incorporated in the ESFAS and instrumentation systems design. NUREG-0737 provides additional guidance on conformance with this requirement.

[21] For conformance with paragraphs (C) and (E) of 10 CFR 50.34(f)(2)(xiv), the reviewer should use the following guidance:

- Ganged reopening of containment isolation valves is not acceptable. Isolation valves should be reopened, valve-by-valve or line-by-line, provided electrical independence and the single-failure criterion for the ESFAS functions continue to be satisfied.

- Containment purge lines and other penetrations that provide a path to the environment should be isolated on a high radiation signal as one of the diverse isolation functions.

NUREG-0737 provides additional guidance on conformance with this requirement.

[22] The review of 10 CFR 50.34(f)(2)(xx) of power supplies is part of Chapter 8, titled “Electric Power,” and it is not reviewed in Chapter 7. The power supplies should conform with the guidance of NUREG-0737.

[23] The review of 10 CFR 50.62 should be coordinated with the organization responsible for the review of reactor systems, which evaluates whether the ATWS mitigation protective functions conform to the ATWS analysis referenced in Chapter 15 of the application, for AOOs, and to verify the adequacy of the design of mechanical systems used to mitigate ATWS.

[24] For the review of 10 CFR 50.49, the staff will coordinate with the organization responsible for the review of equipment qualification, which reviews mild environment qualification, including electromagnetic interference qualification of safety system I&C equipment, instrument sensing lines, lightning protection, and qualification for harsh environments.

[25] The evaluation of the habitability aspects of GDC 19 with respect to radiation protection is addressed in the review of DSRS Chapter 6.

[26] The adequacy of the human factor aspects of the control room design is addressed in the review of DSRS Chapter 18.

[27] The staff should review the ITAAC in Chapter 14 and Tier 1 of the DCD to ensure that the design commitments described in Chapter 7 that demonstrate compliance to regulatory requirements are adequately captured in the ITAAC tables and appropriate verification method(s) (i.e., inspection, test, or analysis), and acceptance criteria are specified.