Draft for Comment

U.S. NUCLEAR REGULATORY COMMISSION

DESIGN-SPECIFIC REVIEW STANDARD FOR NuScale SMR DESIGN

8.4 STATION BLACKOUT

REVIEW RESPONSIBILITIES

Primary - Organization responsible for the electrical engineering review

Secondary - None

I. AREAS OF REVIEW

The term “station blackout“ (SBO) refers to the complete loss of alternating current (ac) electric power to the essential and nonessential switchgear buses in a nuclear power plant (NPP). An SBO, therefore, involves the loss of the offsite electric power system (referred to in industry standards and regulatory guides (RGs) as the ”preferred power system“ concurrent with a turbine trip and unavailability of any emergency ac (EAC) power system. An SBO does not include the loss of available ac power to buses fed by station batteries through inverters or by any alternate ac (AAC) sources specifically provided for SBO mitigation.

The information presented in the safety analysis report (SAR) should be sufficient to support the conclusion that the plant is capable of withstanding and recovering from a complete loss of ac electric power to the essential and nonessential switchgear buses for a minimum of 72 hours. The staff will perform the review to ensure conformance with the requirements of Title 10 of the *Code of Federal Regulations* (CFR), Section 50.63, 10 CFR 50.65, and General Design Criteria (GDCs) 17 and 18 in Appendix A to 10 CFR Part 50, by verifying that the licensee is implementing the relevant guidance of RG 1.155, as supplemented by the guidance and criteria herein.

The analyses performed to demonstrate compliance with 10 CFR 50.63 should remain valid for the life of the NPP. Therefore, if the underlying assumptions change during the life of the NPP, licensees are expected to reevaluate the specified coping duration for their NPPs and the accompanying coping analyses using RG 1.155 or NUMARC-8700, Revision 0, as endorsed by RG 1.155.

The NuScale design, as presented to the staff in oublic familiarization meetings prior to docketing a Design Certification (DC) application, is intended to be capable of coping with a loss of all ac power and resulting plant trip. Therefore, the design is not intended to rely on special SBO mitigation provisions, such as those discussed above, to meet 10 CFR 50.63.” For the Nuscale design, this scenario simply becomes a loss of offsite power event. Unlike other passive designs already certified by the staff, Nuscale does not rely on onsite batteries to power the cool down, but rather relies solely on natural circulation for the duration of the event. The only power required for this event is for the instrumentation to monitor reactor conditions.

The following portions of this design specific review standard (DSRS) provide guidance for the

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staff to evaluate the NuScale design as presented. If additional guidance is needed, the reviewer is directed to the mPower DSRS for more traditional guidance in reviewing SBO for a small modular passive design.

The specific areas of review are as follows:

1. SBO Coping Capability. The review should determine that the capability to achieve and maintain safe-shutdown and containment integrity (non-design-basis accident (DBA)) during an SBO conforms to the guidance provided in Section C.3.2 of RG 1.155. The review should also ensure that any appropriate procedures and training have been developed to implement this capability, including long-term actions following an extended loss of preferred power beyond 72 hours.

2. AAC Power Sources. Passive plant designs need not include an AAC power source if it can be demonstrated that all safety-related functions can be performed without reliance on ac power for 72 hours after the initiating event and the applicant has implemented, as appropriate and necessary, a regulatory treatment of non-safety system (RTNSS) process that conforms to Standard Review Plan (SRP) Chapter 19.3.

3. Procedures and Training. The review should determine that procedures and training conform to the guidance in Sections C.1.3, C.2, and C.3.4 of RG 1.155. Procedures and training should address all operator actions necessary to restore/activate ac power and assure normal long-term core cooling/decay heat removal.

   The review should determine that communication agreements and protocols between the plant and its transmission system operator provide assurance that the NPP operator will be kept aware of (1) changes in the plant switchyard and offsite power grid and (2) local power sources and transmission paths that could be made available to resupply the plant following a loss of offsite power (LOOP) (Reference 15).

4. Quality Assurance (QA) and Specifications for Nonsafety-Related Equipment. The review should determine that QA activities and specifications for nonsafety-related equipment used to meet the requirements of 10 CFR 50.63 conform to the recommendations in Section C.3.5 and Appendix A to RG 1.155. The review should also determine that systems and equipment used to meet the requirements of 10 CFR 50.63 conform to the system and station equipment specification recommendations of Appendix A to RG 1.155. (The reviewer should note that this equipment may not be electrical.)

5. 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 SSCs related to this DSRS section in accordance with DSRS Section 14.3.6, “Electrical System – ITAAC”, and 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 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 DSRS Section 14.3.6 and SRP Section 14.3.

6. <u>COL Action Items and Certification Requirements and Restrictions</u>. For the NuScale DC application, the review will also address COL action items, as well as requirements and restrictions (e.g., interface requirements and site parameters).

   For subsequent COL applications referencing the NuScale DC, COL applicants must address COL action items included in the NuScale DC. Additionally, COL applicants must address requirements and restrictions (e.g., interface requirements and site parameters) included in the DC.

<u>Review Interfaces</u>

Other SRP and DSRS sections interface with this section as follows:

1. The adequacy of the onsite power system is reviewed by the organization responsible for electrical engineering as part of its primary review responsibility for DSRS Sections 8.3.1 and 8.3.2.

2. The adequacy of the offsite power system is reviewed by the organization responsible for electrical engineering as part of its primary review responsibility for DSRS Section 8.2.

3. The organization responsible for the review of DSRS Sections 4.6, 5.4.7, 9.3.6, and 6.3 determines those system components, if any, needing electric power as a function of time for each mode of reactor operation entering an SBO event.

4. The organization responsible for the review of DSRS Sections 9.1.3, 9.2.2, 9.2.6, 9.4.1 through 9.4.4, and 10.4.7, and SRP Sections 9.1.4, 9.2.1, 9.2.4, 9.2.5, 9.3.1, 9.3.3, and 9.5.1 determines those system components, if any, needing electric power as a function of time for each mode of reactor operation entering an SBO event. The review will also verifies, on request, the design adequacy and capability of the identified systems and equipment needed to cope with an SBO for the required duration and recovery period.

5. The organization responsible for the review of DSRS Chapter 7 determines those system components needing electric power as a function of time for each mode of reactor operation and accident condition and, upon request, also verifies the adequacy of the instrumentation and controls used to cope with and recover from an SBO condition.

6. The organization responsible for the review of DSRS Section 16.0 coordinates and performs reviews of technical specifications.

7. The organization responsible for the review of DSRS Sections 6.2.2, 6.2.4 and 6.2.5 determines those system components, if any, needing electric power as a function of time for each mode of reactor operation entering an SBO event.

8. The organization responsible for the review of DSRS Section 14.2 determines the acceptability of the preoperational and initial startup tests and programs.

9. The organization responsible for the review of SRP Sections 13.5.2.2 evaluates the adequacy of administrative, maintenance, testing, and operating procedure programs. In addition, on request, the organization responsible for SRP Sections 13.5.1.1 and

13.5.2.1 reviews potential habitability concerns for those areas that would need operator access during the SBO and recovery period.

10. The organization responsible for the review of SRP Chapter 17 evaluates the design, construction, and operations phases of QA programs, including the general methods for addressing periodic testing and RTNSS in passive plant designs. In addition, while conducting regulatory audits in accordance with Office Instructions NRR-LIC-111 or NRO-REG-108, “Regulatory Audits,” the technical staff may identify quality-related issues. If this occurs, the technical staff should contact the organization responsible for quality assurance to determine if an inspection should be conducted

11. Review of RTNSS is coordinated and performed under SRP Chapter 19 that provides the probabilistic risk assessment for potential risk significance of SSCs.

II. ACCEPTANCE CRITERIA

Requirements

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

1. General Design Criteria (GDC) 17, Electric Power Systems.

2. GDC 18, Inspection and Testing of Electric Power Systems.

3. 10 CFR 50.63, as it relates to the capability to withstand and recover from an SBO.

4. 10 CFR 50.65(a)(4), as it relates to the assessment and management of the increase in risk that may result from proposed maintenance activities before performing the maintenance activities. These activities include, but are not limited to, surveillances, post-maintenance testing, and corrective and preventive maintenance. Compliance with the maintenance rule, including verification that appropriate maintenance activities are covered therein, is reviewed under SRP Chapter 17. Programs for incorporation of requirements into appropriate procedures are reviewed under SRP Chapter 13.

5. 10 CFR 52.47(b)(1), which requires that a DC application contain the proposed 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 facility that incorporates the DC has been constructed and will be operated in conformity with the DC, the provisions of the Atomic Energy Act (AEA), and the NRC's regulations;

6. 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 COL, the provisions of the AEA, and the NRC's regulations.

DSRS Acceptance Criteria

Specific DSRS acceptance criteria acceptable to meet the relevant requirements of the NRC'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.

1. The guidelines of RG 1.155, as they relate to compliance to 10 CFR 50.63 for the NuScale design. NUMARC-8700, Revision 0, also provides guidance acceptable to the staff for meeting these requirements. Table 1 of RG 1.155 provides a cross-reference to NUMARC-8700, Revision 0, and notes when the RG takes precedence.

2. The guidelines and criteria of SECY-90-016 and SECY-94-084 (Reference 23 and 25), as they relate to the use of AAC power sources and RTNSS at plants provided with passive safety systems.

3. The guidelines of RG 1.75 (Reference 6), as they relate to the independence of SBO-related power sources/distribution systems between the onsite and offsite ac power systems. Especially the isolation capability of the battery chargers for the dc system.

Technical Rationale

The technical rationale for application of these acceptance criteria to the areas of review addressed by this DSRS section is discussed in the following paragraphs:

1. Compliance with GDC 17 with respect to SBO focuses on the independence of the SBO-related power sources and their capacity/capability to perform their required functions. GDC 17 assures independence by requiring the inclusion of provisions to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies. For the NuScale passive design, this independence is provided by the battery chargers for the dc power system.

   Meeting the independence and capacity/capability requirements of GDC 17 provides assurance that a reliable electric power supply will be provided to fully respond to an SBO.

2. Compliance with GDC 18 with respect to SBO requires that electric power systems important to safety be designed to permit appropriate periodic inspection and testing of key areas and features to assess their continuity and the condition of their components. These systems shall be designed to test periodically the operability and functional performance of the components of the systems.

   Meeting the requirements of GDC 18 provides assurance that, when necessary, onsite power systems can be appropriately and unobtrusively accessed for required periodic inspection and testing, enabling verification of important system parameters, performance characteristics, and features and detection of degradation and/or impending failure under controlled conditions.

3. Compliance with 10 CFR 50.63 requires that each light-water-cooled NPP be able to withstand and recover from an SBO of specified duration. As required by 10 CFR 50.63, electrical systems must be of sufficient capacity and capability to ensure that the core is cooled and that appropriate containment integrity is maintained in the event of an SBO.

   The capacity of any onsite dc sources required for SBO response must be verified as adequate to address the worst-case SBO load profile and specified duration. For new advanced light-water reactor (ALWR) applications such as NuScale, that use passive safety systems and do not include a spare, full-capacity alternate ac power source for coping with an SBO, the reviewer should ensure that (1) all safety-related functions can be performed without relying on ac power for 72 hours after the initiating event, and (2) the applicant has implemented, as appropriate, an RTNSS process that conforms to Chapter C.IV.9 of RG 1.206. For COL applicants who reference the NuScale certified design, that application will address the implementation of the RTNSS process. RG 1.155 and DSRS Section 8.4 describe guidance acceptable to the staff for meeting the requirements of 10 CFR 50.63 as applied to the unique features of the NuScale design, as presented.

   Meeting the requirements of 10 CFR 50.63 provides assurance that nuclear power plants will be able to withstand or cope with, and recover from, an SBO by providing capability for maintaining core cooling and an appropriate level of containment integrity. The SBO coping capability is reviewed in Chapter 15 of the DSRS.

III. REVIEW PROCEDURES

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

1. Selected Programs and Guidance - In accordance with the guidance in NUREG-0800, "Introduction - Part 2: Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: Integral Pressurized Water Reactor Edition" (NUREG-0800 Intro Part 2) as applied to this DSRS Section, the staff will review the information proposed by the applicant to evaluate whether it meets the acceptance criteria described in Subsection II of this DSRS. As noted in NUREG-0800 Intro Part 2, the NRC requirements that must be met by an SSC do not change under the SMR framework. Using the graded approach described in NUREG-0800 Intro Part 2, the NRC staff may determine that, for certain structures, systems, and components (SSCs), the applicant's basis for compliance with other selected NRC requirements may help demonstrate satisfaction of the applicable acceptance criteria for that SSC in lieu of detailed independent analyses. The design-basis capabilities of specific SSCs would be verified where applicable as part of completion of the applicable ITAAC. The use of the selected programs to augment or replace traditional review procedures is described in Figure 1 of NUREG-0800, Introduction - Part 2. Examples of such programs that may be relevant to the graded approach for these SSCs include:

- 10 CFR Part 50, Appendix A, General Design Criteria (GDC), Overall Requirements, Criteria 1 through 5
- 10 CFR Part 50, Appendix B, Quality Assurance (QA) Program
- 10 CFR 50.49, Environmental Qualification of Electrical Equipment (EQ) Program
- 10 CFR 50.55a, Code Design, Inservice Inspection and Inservice Testing (ISI/IST) Programs
- 10 CFR 50.65, Maintenance Rule requirements
- Reliability Assurance Program (RAP)
- 10 CFR 50.36, Technical Specifications
- Availability Controls for SSCs Subject to Regulatory Treatment of Non-Safety Systems (RTNSS)
- Initial Test Program (ITP)
- Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC)

This list of examples is not intended to be all-inclusive. It is the responsibility of the technical reviewers to determine whether the information in the application, including the degree to which the applicant seeks to rely on such selected programs and guidance, demonstrates that all acceptance criteria have been met to support the safety finding for a particular SSC.

2. In accordance with 10 CFR 52.47(a)(8),(21), and (22), and 10 CFR 52.79(a)(17), (20) and (37), for design certification or combined license applications submitted under Part 52, the applicant is required to (1) address the proposed technical resolution of unresolved safety issues and medium- and high-priority generic safety issues which are identified in the version of NUREG 0933 current on the date up to 6 months before the docket date of the application and which are technically relevant to the design; (2) demonstrate how the operating experience insights have been incorporated into the plant design; and, (3) provide information necessary to demonstrate compliance with any technically relevant portions of the Three Mile Island requirements set forth in 10 CFR 50.34(f), except paragraphs (f)(1)(xii), (f)(2)(ix), and (f)(3)(v) for a DC application, and except paragraphs (f)(1)(xii), (f)(2)(ix), (f)(2)(xxv), and (f)(3)(v) for a COL application. 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 (SER) section.

3. GDC 17 places requirements on the offsite power system and the onsite ac and dc power systems. With respect to SBO for the NuScale design, as presented, only dc power for instrumentation needs to be available. Therefore, the GDC 17 specific requirements that pertain to this DSRS section are (1) independence of the SBO-related power sources and (2) their capability/capacity to perform their intended functions. Independence of the dc power system from the ac systems can be achieved by battery chargers. If so utilized, the battery chargers must be fully qualified as isolation devices in accordance with RG 1.75.

4. GDC 18 places testability requirements on all of the SBO-related power sources. For the dc power system, this verification is performed under DSRS Section 8.3.2.

5. The SBO rule (10 CFR 50.63) requires each plant to demonstrate the capability to withstand (cope with) and recover from an SBO condition lasting for a specified duration (coping duration). Specifically, applicants should do the following:

   A. Establish the duration of an SBO that the plant will be able to withstand (coping duration). For passive designs, this should be a minimum of 72 hours without operator intervention.

   B. Evaluate the plant's capability to withstand and recover from an SBO (coping capability).

   C. Develop the necessary procedures and training to cope with and recover from an SBO.

6. To ensure that the requirements of 10 CFR 50.63 are satisfied, the staff should take the following review steps:

   A. <u>SBO Coping Duration</u>. 10 CFR 50.63(a)(1) and 10 CFR 50.63(c)(1)(i) requires each nuclear power plant to specify an SBO coping duration based on site- and plant-specific factors that contribute to the likelihood of and capability for restoring alternating current power (ac) following of an SBO. These factors include consideration for redundancy and reliability of onsite emergency ac power sources. Since passive plants do not have EAC power sources, passive plant designs meet the 10 CFR 50.63 for specifying a coping duration by demonstrating that safety-related functions are assured for a minimum of 72 hours following an SBO event. The 72 hour coping duration for passive plant designs is consistent with the station blackout duration approved by NRC staff, as reflected in SECY-90-016 and SECY-94-084 (Reference 23 and 25).

   B. <u>SBO Coping Capability</u>. The staff will review the DCD to determine that the capability to cope with an SBO lasting for 72 hours conforms to the guidance in Section C.3.2 of RG 1.155. The review should ensure that the capability to maintain adequate core cooling and appropriate containment integrity for this specified coping duration is adequately demonstrated and appropriate procedures and training are implemented to withstand (cope with) the event. Passive designs employ an ac-independent approach to cope with SBO. The plant relies on available sources of energy that are independent of ac power. Therefore, the reviewer should determine that an analysis conforming to the guidance in Sections C.3.2.1 to C.3.2.4 of RG 1.155 demonstrates the capability to achieve and maintain safe-shutdown for a minimum of 72 hours without operator intervention.

7. The reviewer should verify that the applicant's determination of the plant's ability to cope with an SBO should be based on the following general criteria, initial conditions and baseline assumptions (References 7 and 38):

   A. Because of the presence of substantial decay heat, events initiated from 100-percent power bound the potential for core damage from an SBO. Therefore, the coping analysis should be performed assuming that the SBO event occurs while the reactor is operating at 100-percent rated thermal power and has been at this power level for at least 100 days.

B. Immediately before the postulated SBO event, the reactor and supporting systems are within normal operating ranges for pressure, temperature, and water level. All plant equipment is either normally operating or available from the standby state.

C. It is assumed that a reasonable set of operator actions will occur to mitigate the effects of an SBO and recover from the event. Operator actions are assumed to follow plant operating procedures for the underlying symptoms or identified event scenario associated with an SBO.

D. Actions specified in procedures for SBO are predicated on the use of instrumentation and controls powered by vital buses supplied by station batteries.

E. The dc power needs for SBO may be estimated using Institute for Electrical and Electronics Engineers (IEEE) Standard (Std.) 485 (Reference 14) or an analogous methodology, if appropriate, for the NuScale design.

F. Since the capacity of battery storage varies with electrolyte temperature, calculations should assume the lowest temperature normally expected of the battery.

G. The capability of any/all electrical systems and components necessary to provide core cooling and decay heat removal following an SBO should be identified along with their loading requirements.

H. The ability to maintain adequate reactor coolant system inventory to ensure that the core is cooled should be demonstrated, taking into consideration any possible pathways for inventory to escape.

I. The design adequacy and capability of equipment needed to cope with an SBO for the required duration and recovery period should be addressed and evaluated as appropriate for the associated environmental conditions. This should include consideration of the following:

   i. Potential environmental effects on the operability and reliability of equipment necessary to cope with the SBO, including possible effects of fire protection systems

   ii. Potential habitability concerns for those areas that would need operator access during the SBO and recovery period

J. Equipment will be considered acceptable for SBO temperature environments if an assessment has been performed by the applicant that provides reasonable assurance that the necessary equipment will remain operable.

K. The ability to maintain appropriate containment integrity should be addressed. Appropriate containment integrity for SBO means that adequate containment integrity is ensured by providing the capability, independent of ac power supplies, for valve position indication and closure for any electrically-operated containment isolation valves that may be in the open position at the onset of an SBO. This does not include the following valves:

i. Valves normally locked closed during operation

ii. Valves that fail closed on a loss of power

iii. Check valves

iv. Valves in nonradioactive closed-loop systems not expected to be breached in an SBO (not including any lines that communicate directly with containment atmosphere)

8. AAC Power Sources. For new ALWR plants, the Commission has established a policy (Reference 23) that such plants should have an AAC power source of diverse design and capable of powering at least one complete set of normal shutdown loads. In SECY-94-084 and SECY-95-132 (Reference 25 and 26), the Commission modified these criteria for ALWRs that use passive safety systems. Specifically, an AAC power source is not necessary for passive plant designs that (1) do not need ac power to perform safety-related functions for 72 hours following the onset of an SBO and (2) meet the guidelines in SRP Section 19.3 regarding RTNSS, as applicable to the NuScale design.

9. Procedures and Training. The staff will review the applicant's procedures and training programs to ensure that they conform to the guidance in Sections C.1.3, C.2, and C.3.4 and Appendix B to RG 1.155 and include all operator actions necessary to do the following:

   A. Cope with the occurrence of an SBO occurring during any mode of plant operation. Procedures developed to cope with an SBO should be integrated with the plant-specific technical guidelines and emergency operating procedures developed using the emergency operating procedure upgrade program established in response to Supplement 1 of NUREG-0737. The task analysis portion of the emergency operating procedure upgrade program should include an analysis of instrumentation adequacy during an SBO.

   B. Monitor plant conditions over the first 72 hours and prepare for longer term actions that may be necessary to recharge the batteries.

   C. Restore offsite power sources or use any nearby power sources (which may include nearby or onsite gas turbine generators, portable generators, hydro generators, and black start fossil power plants) in the event of a LOOP.

10. In addition, the reviewer should determine that plant operating procedures developed to respond to an SBO event are consistent with the following general guidelines:

A. Plant operating procedures should identify any sources of potential inventory loss and specify actions to prevent or limit significant loss.

B. Plant operating procedures should specify clear criteria for transferring to the next preferred source of water should such need arise.

C. Plant operating procedures should specify any actions necessary to permit appropriate containment isolation and safe-shutdown valve operations beyond 72 hours if normal ac power is still unavailable.

D. Plant operating procedures should identify any portable lighting necessary for ingress and egress to plant areas containing shutdown equipment requiring manual operation.

E. Plant operating procedures should consider the effects of ac power loss on area access, as well as any need to gain entry to locked areas where remote equipment operation may become necessary.

F. Plant operating procedures should consider the effects of a loss of ac power on communications capabilities, including the potential for a loss of communications with offsite agencies.

G. Plant operating procedures should consider the loss of any heat tracing that would affect equipment necessary to cope with an SBO.

H. Plant operating procedures should contain appropriate communication protocols between the NPP and its transmission system operator (Reference 15). With regard to SBO, these protocols should aid the operator in determining the Availability of local power sources and transmission paths that could be made available to resupply the plant following a LOOP event.

11. <u>QA and Specification Guidance for SBO Equipment that is not Safety-Related</u>. The staff will review QA activities and specifications for nonsafety-related equipment used to meet the requirements of 10 CFR 50.63 to ensure that they conform to the recommendations in Section C.3.5 and Appendix A to RG 1.155. The review should also determine that systems and equipment used to meet the requirements of 10 CFR 50.63 conform to the system and station equipment specification recommendations of Appendix B to RG 1.155. The NRC staff will accept the nonsafety systems identified in Appendix B to RG 1.155 for responding to an SBO.

    For the passive NuScale design, the applicant should define any active systems that are relied upon for defense-in-depth purposes and that are necessary to meet passive ALWR plant safety and investment protection goals. The staff reviews QA controls applicable to the SSCs within the RTNSS process under SRP Section 19.3.

12. 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 both DC and COL applications, DSRS Section 14.3.6 and 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 SER. The reviewer also states the bases for those conclusions.

On the basis of the staff's detailed review and evaluation of the SBO capability described in the SAR for DCD/COL (Facility), the staff concludes that the applicant has appropriately evaluated the facility against the guidelines of RG 1.155 and this DSRS section. The SAR DCD/COL acceptably demonstrates that the plant is in compliance with the applicable provisions of GDCs 17 and 18 and 10 CFR 50.63, as they relate to the capability to achieve and maintain safe-shutdown (non-DBA) in the event of an SBO.

Accordingly, the staff concludes that the plant design is acceptable and meets the requirements of GDCs 17 and 18 of Appendix A to 10 CFR Part 50, as they relate to the requirements of 10 CFR 50.63 and 10 CFR 50.65.

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 DSRS section.

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

V. IMPLEMENTATION

The regulations in 10 CFR 52.17(a)(1)(xii), 10 CFR 52.47(a)(9), and 10 CFR 52.79(a)(41) establish requirements for applications for ESPs, DCs, and COLs, respectively. These regulations require the application to include an evaluation of the site (ESP), standard plant design (DC), or facility (COL) against the Standard Review Plan (SRP) revision in effect six months before the docket date of the application. While the SRP provides generic guidance, the staff developed the SRP guidance based on the staff's experience in reviewing applications for construction permits and operating licenses for large light-water nuclear power reactors. The proposed small modular reactor (SMR) designs, however, differ significantly from large light-water nuclear reactor power plant designs.

In view of the differences between the designs of SMRs and the designs of large light-water power reactors, the Commission issued SRM- COMGBJ-10-0004/COMGEA-10-0001, "Use of Risk Insights to Enhance the Safety Focus of Small Modular Reactor Reviews," dated August

31, 2010 (ML102510405) (SRM). In the SRM, the Commission directed the staff to develop risk-informed licensing review plans for each of the SMR design reviews, including plans for the associated pre-application activities. Accordingly, the staff has developed the content of the DSRS as an alternative method for the evaluation of a NuScale-specific application submitted pursuant to 10 CFR Part 52, and the staff has determined that each application may address the DSRS in lieu of addressing the SRP, with specified exceptions. These exceptions include particular review areas in which the DSRS directs reviewers to consult the SRP and others in which the SRP is used for the review. If an applicant chooses to address the DSRS, the application should identify and describe all differences between the design features (DC and COL applications only), analytical techniques, and procedural measures proposed in an application and the guidance of the applicable DSRS section (or SRP section as specified in the DSRS), and discuss how the proposed alternative provides an acceptable method of complying with the regulations that underlie the DSRS acceptance criteria.

The staff has accepted the content of the DSRS as an alternative method for evaluating whether an application complies with NRC regulations for NuScale SMR applications, provided that the application does not deviate significantly from the design and siting assumptions made by the NRC staff while preparing the DSRS. If the design or siting assumptions in a NuScale application deviate significantly from the design and siting assumptions the staff used in preparing the DSRS, the staff will use the more general guidance in the SRP as specified in 10 CFR 52.17(a)(1)(xii), 10 CFR 52.47(a)(9), or 10 CFR 52.79(a)(41), depending on the type of application. Alternatively, the staff may supplement the DSRS section by adding appropriate criteria in order to address new design or siting assumptions.

VI. REFERENCES

1. 10 CFR 50.2, ”Definitions.”

2. 10 CFR 50.63, ”Loss of All Alternating Current Power.”

3. 10 CFR Part 50, Appendix A, GDC 17, ”Electric Power Systems.”

4. 10 CFR Part 50, Appendix A, GDC 18, ”Inspection and Testing of Electric Power Systems.”

5. 10 CFR Part 52, ”Licenses, Certifications, and Approvals for Nuclear Power Plants.”

6. RG 1.75, “Criteria for Independence of Electrical Safety Systems.”

7. RG 1.155, ”Station Blackout.”

8. RG 1.160, ”Monitoring the Effectiveness of Maintenance at Nuclear Power Plants.”

9. DSRS Section 8.1, “Electric Power – Introduction,” Table 8-1, ”Acceptance Criteria and Guidelines for Electric Power Systems.”

10. DSRS Section 8.2, ”Offsite Power.”

11. DSRS Section 8.3.1, “AC Power Systems (Onsite).”

12. DSRS Section 8.3.2, “DC Power Systems (Onsite).”

13. Intentionally left blank.

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24. SECY-94-084, ”Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs,” March 28, 1994. Approved in the SRM, dated June 30, 1994.

25. SECY-95-132, ”Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems (RTNSS) in Passive Plant Designs.” Approved in the SRM, dated June 28, 1995.

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