Draft for Comment

U.S. NUCLEAR REGULATORY COMMISSION

DESIGN-SPECIFIC REVIEW STANDARD FOR NuScale SMR DESIGN

9.3.6 CONTAINMENT EVACUATION AND FLOODING SYSTEMS

REVIEW RESPONSIBILITIES

Primary - Organization responsible for the review of ventilation and air filtration

Secondary - None

I. AREAS OF REVIEW

NuScale is an integral, pressurized, small modular (SMR) light water reactor with the reactor, steam generator, pressurizer, and control rod drives all located in a single pressure vessel. The NuScale reactor containment is an evacuated, low alloy steel vessel surrounding the smaller reactor vessel and immersed in a bay of a large borated reactor building pool that serves as the passive containment heat dump and ultimate heat sink.

Containment evacuation and flooding systems (CEFS) is a combination of containment evacuation system (CES) and containment flooding and drain system (CFDS). This design-specific review standard (DSRS) is established to review both CES and CFDS.

The function of the containment evacuation system (CEFS) is to establish and maintain a vacuum in the containment vessel, detect and identify reactor coolant system (RCS) leakage, and provide means to flood containment for transfer of decay heat to the ultimate heat sink (UHS) during refueling and beyond-design-basis events.

The responsible staff reviews information regarding the containment evacuation system presented in Section 9.3.6 of the construction permit, design certification (DC), or combined license (COL) application to ensure compliance with the requirements of General Design Criteria (GDC) 1, 2, 4, 5, 52, 53, 54, 56, and 60. The review includes such components as piping, vacuum pumps, isolation valves, discharge separators, heat exchangers, and instrumentation.

The specific areas of review are as follows:

1. The capability of CEFS to establish vacuum in the containment vessel as designed.

2. The capability of CEFS to ensure containment vacuum is maintained at an acceptable value as a starting point at the onset of design basis accident. This acceptable value is enforced by Technical Specifications.

3. The capability of CEFS to detect and identify RCS leakage.

4. The capability of CEFS to provide means to flood containment for transfer of decay heat to the UHS during refueling and beyond-design-basis events.

5. In the system's purge mode, CEFS provides adequate pressurized air to purge liquid out of the containment after refueling and during normal operation.

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Review Interfaces

Other standard review plan (SRP) or DSRS sections interface with this section as follows:

1. SRP Sections 3.2.1 and 3.2.2: review of system seismic design and quality group for valves, respectively.

2. DSRS Sections 3.3.1, 3.3.2, 3.5.3, 3.7.1 through 3.7.3, 3.8.4, and 3.8.5, and SRP Section 3.7.4: review to determine the acceptability of the design analysis, procedures, and criteria to establish the ability of seismic Category I structures housing the system and supporting systems to withstand the effects of natural phenomena like the safe shutdown earthquake, the probable maximum flood, and tornado missiles.

3. DSRS Sections 3.8.2 through 3.8.5: review of the containment isolation structure design for adequate protection against earthquakes.

4. SRP Sections 3.9.1 through 3.9.3: review to determine that the components, piping, and structures are designed in accordance with applicable codes and standards.

5. SRP Section 3.9.6: reviews of the adequacy of the inservice testing program of pumps and valves.

6. DSRS Section 3.11: review of the environmental qualification of safety-related valves.

7. DSRS Sections 6.2.4 and 6.2.6: review of containment isolation system and the overall containment leakage testing program, respectively.

8. DSRS Section 6.6: verification whether inservice inspection requirements are met for system components.

9. DSRS Sections 7, 8.3.1 and 8.3.2: review to determine the adequacy of the design, installation, inspection, and testing of all essential electrical components (sensing, control and power) required for proper operation.

10. DSRS Section 11.3: functional performance ensuring that the system meets acceptable limits for radioactive release during normal operations.

11. DSRS Section 11.5: evaluation of the capability of the system to detect and control leakage of radioactive contamination.

12. DSRS Section 16.0: review of proposed technical specifications for operability and leakage-rate testing of isolation barriers and closure time for containment isolation valves.

13. SRP Section 17.5: review of quality assurance programs.

14. SRP Section 19: review of beyond-design-bases events.

15. SRP Section 19.3: review of regulatory treatment of nonsafety systems for passive advanced light water reactors

II. ACCEPTANCE CRITERIA

Requirements

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

1. General Design Criteria (GDC) 1, as it relates to designing, fabricating, erecting, and testing safety-related SSCs to quality standards commensurate with the importance of the safety functions to be performed.

2. GDC 2, as it relates to the system being capable of withstanding the effects of earthquakes.

3. GDC 4, as it relates to designing safety-related SSCs to accommodate the effects of and to be compatible with environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, and as it relates to the requirement that these SSCs shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids.

4. GDC 5, as it relates to shared systems and components important to safety.

5. GDC 52, as it relates to designing containment isolation valves so that periodic integrated leakage rate testing can be conducted at containment design pressure.

6. GDC 53, as it relates to designing provisions to permit appropriate inspection of important areas (such as primary containment penetrations), an appropriate surveillance program, and leakage rate testing at the containment design pressure of penetrations having resilient seals and expansion bellows.

7. GDC 54, as it relates to designing piping systems penetrating primary reactor containment with a capability to determine if valve leakage rate is within acceptable limits.

8. GDC 56, as it relates to providing isolation valves for each line that connects directly to the containment atmosphere and penetrates ordinary reactor containment as follows:

   A. One locked closed isolation valve inside and one locked closed isolation valve outside containment; or

   B. One automatic isolation valve inside and one locked closed isolation valve outside containment; or

   C. One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or

   D. One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

9. GDC 60, as it relates to the capability to suitably control release of radioactive materials to the environment.

10. 10 CFR 50, App. J, provides guidance for the primary reactor containment leakage testing for water–cooled power reactors.

11. 10 CFR 50.34(f)(2)(xiv), provides additional TMI-related requirements for the design of containment isolation systems.

12. CFR 52.47(b)(1), which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a 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 Nuclear Regulatory Commission's (NRC's) regulations.

13. 10 CFR 52.80(a), as it relates to the requirement that a COL application contain the proposed inspections, tests, 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.

14. 10 CFR 52.47(a)(2), as it relates to the expectation that the standard plant will reflect through its design, construction, and operation an extremely low probability for accidents that could result in the release of radioactive fission products.

15. 10 CFR 52.79(a)(38), as it relates to the requirement that a COL application for a light water reactor design must contain a final safety analysis report (FSAR) that includes a description and analysis of design features for the prevention and mitigation of severe accidents.

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. For GDC 2, acceptance is based on the guidance of RG 1.29, Position C.1, for safety-related portions and Position C.2 for nonsafety-related portions.

2. For GDC 60, acceptance is based on the guidance of RG 1.140 as related to design, testing, inspection, testing, and maintenance criteria for normal atmosphere cleanup systems, ventilation exhaust systems, air filtration, and adsorption units of light-water-cooled nuclear power plants.

3. Policy Statement, "Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants," 50 FR 32138, August 8, 1985. The acceptance is based on whether the design can demonstrate in compliance with the criteria of the current Commission regulations related to severe accidents, including the Three Mile Island requirements for new plants as reflected in the construction permit rule (10 CFR 50.34(f)).

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. GDC 2 as related to the system being capable of withstanding the effects of earthquakes, requires that SSCs important to safety be designed to withstand the effects of a design basis earthquake without loss of capability to perform their safety functions. GDC 2 applies to this DSRS section because the reviewer evaluates the containment isolation valves in the CEFS for the capability to isolate the containment. This requirement ensures that in the event of a design-basis earthquake, these containment isolation valves will remain functional and the failure of any nonessential portion of the system or of other systems not designed to seismic Category I standards will not result in offsite doses in excess of 5 mSv (0.5 rem) to the whole body or an equivalent dose to any part of the body.

2. GDC 60 requires provisions to be installed in the nuclear power unit design to ensure suitable controls on the release of radioactive materials in gaseous effluents during normal reactor operation, including anticipated operational occurrences.

   GDC 60 requirements apply to the design of the CEFS because one of its functions, combining with the design of Gaseous Rad-Waste Management System, is to control the quantities of radioactive material in gaseous effluents released to the environment. RG 1.140 provides design, testing, and maintenance criteria acceptable to the staff for air filtration and absorption units of normal ventilation exhaust systems in light-water-cooled nuclear power plants.

   Meeting the GDC 60 requirements provides assurance that release of radioactive materials entrained in gaseous effluents will not exceed the limits specified in 10 CFR Part 20 for normal operation and anticipated operational occurrences.

3. For severe accident evaluation as related to Policy Statement 50 FR 32138, the staff reviews the nuclear power plant design with a conclusion of safety acceptability using an approach that stresses deterministic engineering analysis and judgment complemented by Probabilistic Risk Assessment (PRA).

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. Upon request from the primary review organization, the review organizations with review interface responsibilities, as noted in Subsection I, will provide input for the areas of review, as stated in Subsection I of this DSRS section. The input obtained will ensure that the review is complete. The primary review organization ensures that the design and functional capability of the containment evacuation system conforms to the requirements of Policy Statement 50 FR 32138 and GDC 2, 5, and 60.

   The review should concentrate on the following areas:

A. The designed CEFS containment isolation valves are installed only outside the containment. The review guidelines discussed in DSRS 6.2.4 “Containment Isolation System” will be followed to determine its design adequacy.

B. The CEFS will discharge liquid rad-waste to Liquid Rad-Waste Management System. The review guidelines discussed in DSRS 11.2 “Liquid Rad-Waste Management System” and the applicable portion of DSRS Section 11.5 “Evaluation of the Capability of the System to Detect and Control Leakage of Radioactive Contamination” will be followed to determine its design adequacy.

C. The CEFS will discharge gaseous rad-waste to Gaseous Rad-Waste Management System. The review guidelines discussed in DSRS 11.3 “Gaseous Red-Waste Management System” and the applicable portion of DSRS 11.5 “Evaluation of the Capability of the System to Detect and Control Leakage of Radioactive Contamination” will be followed to determine its design adequacy.

D. The CEFS will flood containment during beyond-design basis events. The review guidelines discussed in SRP 19 “Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors,” will be followed to determine its design adequacy.

E. The primary review organization reviews the capability of CEFS to detect and identify RCS leakage. The staff should verify that the designed instrumentation can monitor containment vessel space for radiation and temperature such that RCS leakage can be detected and identified during normal operation.

F. The primary review organization reviews the capability of CEFS to ensure containment vacuum can be maintained at an acceptable value. The staff should verify that vacuum pumps are sized to be able to draw vacuum in the containment to the designed value within the time frame specified in the FSAR.

G. The primary review organization reviews the capability of CEFS to flood containment for transfer of decay heat to UHS during refueling mode. The staff should verify that the system design and equipment capacity provided in the applicant’s technical submittal are adequate to flood containment for the transfer of decay heat to the UHS during refueling operation.

H. The primary review organization reviews the design features and component capacities of CEFS to flood containment for transfer of decay heat to UHS during beyond-design-basis events. For severe accident evaluation, the reviewer should follow Policy Statement 50 FR 32138, B.2.d “Completion of a staff review of the design with a conclusion of safety acceptability using an approach that stresses deterministic engineering analysis and judgment complemented by PRA.” The staff may perform engineering analysis based on CEFS design features. This engineering analysis is complemented by PRA with a conclusion of safety acceptability to assure that adequate decay heat can be transferred to UHS during beyond-design-basis events.

   SRP section 19 also pertains to the staff review of the applicant’s deterministic evaluation of design features for the prevention or mitigation of severe accidents.

The staff reviews the applicant's assumptions related to equipment availability and compares them to Technical Specifications requirements. Risk-significant equipment should be evaluated with respect to 10 CFR 50.36(c)(2)(ii)(D) to determine whether additional Technical Specifications requirements are needed. The staff may also review the results of sensitivity studies performed to demonstrate the risk benefit of equipment that is controlled only by administrative controls (e.g., maintenance rule implementation).

As expressed in the review interfaces portion of Subsection I of this DSRS section, other technical branches are expected to use the applicant's PRA results and insights to inform their review based on risk significance.

The staff reviews the PRA-related ITAAC, COL action items, and other commitments, including any actions identified or proposed to address them. The applicant's PRA-based insights table should also identify these items. The staff should note any item that cannot be resolved until after the COL application phase and review the commitments and schedule for resolution of the given items, and the proposed method of completion.

6. For passive plant designs, the staff reviews the applicant's use of the PRA to identify non-safety systems, structures and components that require regulatory treatment (i.e., to support the regulatory treatment of non-safety systems (RTNSS) program). The primary review organization reviews to determine whether CEFS is relied on to ensure long-term safety for the period beginning 72 hours after a design basis event, e.g., RTNSS Category B. If RTNSS Category B is considered applicable, CEFS should be reviewed to assure that the applicable RTNSS program outlined in SRP 19.3 is followed.

7. 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 FSAR meets the acceptance criteria. DCs have referred to the FSAR as the design control document. The reviewer should also consider the appropriateness of identified COL action items. The reviewer may identify additional COL action items; however, to ensure these COL action items are addressed during a COL application, they should be added to the DC FSAR.

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

   For review of both DC and COL applications, DSRS 14.3.7 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 staff's technical review and analysis, as augmented by the application of programmatic requirements in accordance with the staff's technical review approach in the DSRS Introduction, support conclusions of the following type to be included in the staff's safety evaluation report. The reviewer also states the bases for those conclusions.

To support the basis for the staff's acceptance of the CEFS system, the reviewer of the CEFS should include in the staff's safety evaluation report (SER), as necessary, the results of the

reviews for the five DSRS sections above. The SER write-up should demonstrate conformance with the Commission regulations in the manner indicated. The staff concludes consistent with Policy Statement 50 FR 32138 that the CES functional design is acceptable and meets the requirements of General Design Criteria 2, 5, 60. The conclusion is based on the following: [The reviewer should discuss each item of the regulations or related set of regulations as indicated.]

1. The applicant has met the requirements of (cite regulation) with respect to (state limits of review in relation to regulation) by (for each item that is applicable to the review, state how it was met and why it is acceptable with respect to regulation being discussed):

   A. meeting the regulatory positions in Regulatory Guide ________ or Guides;

   B. providing and meeting an alternative method to regulatory positions in Regulatory Guide ________, that the staff has reviewed and found to be acceptable;

   C. meeting the regulatory position in BTP;

   D. using calculational methods for (state what was evaluated) that have previously been reviewed by the staff and found acceptable; the staff has reviewed the impact parameters in this case and found them to be suitably conservative or performed independent calculations to verify acceptability of their analysis; and/or

   E. meeting the provisions of (industry standard number and title) that has been reviewed by the staff and determined to be appropriate for this application.

2. Repeat discussion for each regulation cited above.

   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. <u>IMPLEMENTATION</u>

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. Policy Statement, "Severe Reactor Accidents Regarding Future Designs and Existing Plants," 50 FR 32138, August 8, 1985.

2. SRP Section 3.2.1, Seismic Classification, Rev 2.

3. SRP Section 3.2.2, System Quality Group Classification, Rev 2.

4. SRP Section 3.9.1, Special Topics for Mechanical Components, Rev 3.

5. SRP Section 3.9.2, Dynamic Testing and Analysis of Systems, Structures, and Components, Rev 3.

6. SRP Section 3.9.3, ASME Code Class 1, 2, and 3 Components, and Component Supports, and Core Support Structures, Rev 2.

7. SRP Section 3.9.6, Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints, Rev 3.

8. DSRS Section 6.2.4, Containment Isolation System, Draft Rev 0.

9. DSRS Section 6.2.6, Containment Leakage Testing, Draft Rev 0.

10. DSRS Section 6.6, Inservice Inspection and Testing of Class 2 and 3 Components, Rev 2.

11. DSRS Section 7.7, Control Systems, Rev 5.

12. DSRS Section 8.3.1, A-C Power Systems (Onsite), Draft Rev 0.

13. DSRS Section 11.2, Liquid Rad-Waste System, Draft Rev 0.

14. DSRS Section 11.3, Gaseous Rad-Waste Management System, Draft Rev 0.

15. DSRS Section 11.5, Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems, Draft Rev 0.

16. DSRS Section 14.3.5, Instrumentation and Controls - Inspections, Tests, Analyses, and Acceptance Criteria, Draft Rev 0.

17. DSRS Section 14.3.7, Plant Systems - Inspections, Tests, Analyses, and Acceptance Criteria, Draft Rev 0.

18. DSRS Section 16.0, Technical Specifications, Draft Rev 0.

19. SRP Section 17.1, Quality Assurance During the Design and Construction Phases, Rev 2.

20. SRP Section 19.0, Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors, Draft Rev 3.

21. SRP Section 19.1, Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities, Rev 3.

22. SRP 19.3, Regulatory Treatment of Non-Safety Systems (RTNSS) for Passive Advanced Light Water Reactors, Draft Rev 0.

23. 10 CFR 52.47(b)(1).

24. 10 CFR 52.47(a)(2).

25. 10 CFR 52.79(a)(38).