U.S. NUCLEAR REGULATORY COMMISSION

DESIGN-SPECIFIC REVIEW STANDARD for NuScale SMR DESIGN

6.2.6 CONTAINMENT LEAKAGE TESTING

REVIEW RESPONSIBILITIES

Primary - Organization responsible for the review of NuScale SMR containment integrity.

Secondary - None

I. AREAS OF REVIEW

NuScale is an integral, pressurized water, small modular reactor (SMR) with the reactor, steam generator, pressurizer, and control rod drives all 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 large, borated reactor building pool that serves as the passive ultimate heat sink for containment heat removal.

The description of the reactor containment leakage rate testing program is reviewed for conformance to Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors"; and General Design Criteria (GDCs) 52, "Capability for Containment Leakage Rate Testing," 53, "Provisions for Containment Testing and Inspection," and 54, "Systems Penetrating Containment."

Appendix J includes two options, A and B, either of which can be chosen by an applicant or licensee for meeting the requirements of the appendix. Option A is "Prescriptive Requirements" and Option B is "Performance-Based Requirements." In accordance with Option B, Section V., an applicant or licensee may choose to comply with all of Option A, all of Option B, or one of the options for containment integrated leakage rate tests (CILRTs) and the other option for local leakage rate tests (LLRTs). If a mixed approach is used, experience indicates it will likely be Option B for CILRTs and Option A for LLRTs, because of the much longer CILRT interval available under Option B. Regardless of the choice made, a plant's technical specifications (TS) will indicate the choice, and a subsequent change in choice would be implemented through a TS change.

Despite the differences between Option A and Option B, there are many similarities and the review guidance below will apply to either option, unless otherwise stated.

The specific areas of review are as follows:

1. CILRTs (Type A tests as defined by Appendix J), including pretest requirements, general test methods, acceptance criteria for preoperational and periodic leakage rate tests, provisions for additional testing in the event of failure to meet acceptance criteria, and scheduling of tests

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2. containment penetration leakage rate tests (Type B tests as defined by Appendix J), including identification of containment penetrations, general test methods, test pressures, acceptance criteria, and scheduling of tests

3. containment isolation valve leakage rate tests (Type C tests as defined by Appendix J), including identification of isolation valves, general test methods, test pressures, acceptance criteria, and scheduling of tests

4. TS pertaining to containment leakage rate testing are reviewed at the combined license (COL) stage, or, in some cases, as part of the design certification (DC) review under 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants"

5. Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC)

   A. For DC and COL reviews, the staff reviews the applicant's proposed ITAAC associated with the structures, systems, and components (SSCs) related to this design-specific review standard (DSRS) section in accordance with Standard Review Plan (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 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 SRP Section 14.3.

6. COL Action Items and Certification Requirements and Restrictions

   A. For a DC application, the review will also address COL action items and requirements and restrictions (e.g., interface requirements and site parameters).

   B. For a COL application referencing a DC, a COL applicant must address COL action items (referred to as COL license information in certain DCs) included in the referenced DC. Additionally, a COL applicant must address requirements and restrictions (e.g., interface requirements and site parameters) included in the referenced DC.

7. Operational Program Description and Implementation

   A. For a COL application, the staff reviews the Containment Leak Rate Testing program description and the proposed implementation milestones. The staff also reviews Final Safety Analysis Report (FSAR) Table 13.x to ensure that the Containment Leak Rate Testing program and associated milestones are included. Specific to this DSRS section are the Containment leak Rate Testing program based on the requirements of 10 CFR Part 50, Appendix J.

<u>Review Interfaces</u>

Other SRP sections interface with this section as follows:

1. For COL reviews of operational programs, the review of the applicant's implementation plan is performed under SRP Section 13.4, "Operational Programs."

2. The review of the probabilistic risk assessment is performed under SRP Section 19.3 for potential risk significance of SSCs.

II. ACCEPTANCE CRITERIA

Requirements

Acceptance criteria are based on meeting the relevant requirements of the following U.S. Nuclear Regulatory Commission (NRC) regulations:

Conformance with the requirements of Option A of Appendix J, or the requirements of Option B of Appendix J and the provisions of Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program," constitutes an acceptable basis for satisfying the requirements of the following GDCs applicable to containment leakage rate testing:

1. GDC 52, as it relates to the reactor containment and exposed equipment being designed to accommodate the test conditions for the CILRT (up to the containment design pressure)

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

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

4. GDC 5, "Containment Design Basis," as it relates to providing assurance that sharing of SSCs important to safety among nuclear power units will not significantly impair their ability to perform their safety functions

5. 10 CFR 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance," as it relates to an applicant assuming the expected demonstrable leakage rate from the containment, as an aid in evaluating a proposed nuclear power plant site

    A. Nuclear power plant leakage rate testing experience shows a design leakage rate of 0.1 percent per day provides adequate margin above typically measured containment leakage rates and is compatible with current leakage rate test methods and test acceptance criteria. Therefore, the minimum acceptable design containment leakage rate should not be less than 0.1 percent per day.

6. the reactor containment leakage rate testing program, as described in the safety analysis report (SAR) or design certification document (DCD), will be acceptable if:

    A. Under Option A, the program meets the requirements stated in Option A of Appendix J to 10 CFR Part 50. Appendix J, Option A, provides the test requirements and acceptance criteria for preoperational and periodic leakage rate testing of the reactor containment and of systems and components that penetrate the containment. Exemption from Appendix J requirements will be reviewed on a case-by-case basis.

B. Under Option B, the program meets the requirements stated in Option B of Appendix J to 10 CFR Part 50 and, under Sections V.B.2 and V.B.3 of Option B, either complies with methods approved by the Commission and endorsed in a RG (RG 1.163) and includes a requirement to do so in the TS, or complies with the provisions of some other implementation document that has been adequately justified to the staff, with supporting analyses, and is cited as a requirement in the TS.

7. 10 CFR 52.47(b)(1), which requires a DC application to contain the proposed ITAAC 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 design certification, the provisions of the Atomic Energy Act (AEA), and the NRC's regulations

8. 10 CFR 52.80(a), which requires a COL application to 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 that 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 (for 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.

Appendix J, Option A, Section III.A.1(a), requires no repairs or adjustments be made to the containment before the performance of the CILRT so the containment can be tested in as close to the "as is" condition as practical. Under Option B, RG 1.163 endorses Nuclear Energy Institute (NEI) 94-01, Revision 0 (with certain exceptions), which provides similar guidance in Sections 8.0 and 9.0. Instrumentation lines that penetrate containment; however, are sometimes isolated for the CILRT. To ensure that they are included in the test, the following should be done. Leakage rate testing of instrumentation lines that penetrate containment may be done in conjunction with either the LLRTs or the CILRT. Instrumentation lines that are not locally leakage rate tested should not be isolated from the containment atmosphere during the performance of the CILRT. The measured leakage rates from instrumentation lines that are locally leakage rate tested, and also isolated during CILRTs, should be added to the CILRT result. Provisions should be made to ensure that instrumentation lines isolated during the CILRT are restored to their operable status following the test.

All leakage rate tests, performed by either pneumatic or hydrostatic means, should have the capability to quantify the leakage rates either explicitly or by a conservative bounding method to satisfy test acceptance criteria in Appendix J and the TS.

Appendix J, Option A, Section III.C.1, prescribes methods for conducting the containment isolation valve leakage rate tests. Under Option B, RG 1.163 endorses NEI 94-01, Revision 0 (with certain exceptions), which provides similar guidance in Sections 8.0 and 10.0. At the standard DC stage, the applicant should identify all containment isolation valves that will be locally (Type C) leakage rate tested with the test pressure applied in a direction opposite to that which would occur under accident conditions and should commit to justify, at the COL stage, that such testing will result in equivalent or more conservative results.

NEI 94-01, Revision 0 (Section 6.0), and ANSI/ANS-56.8-1994 (Section 3.3.1) state that Type B or Type C tests are not required for the following cases:

1. containment boundaries that do not constitute potential containment atmospheric leakage pathways during and following a design-basis loss-of-coolant accident (DB LOCA)

2. containment boundaries sealed with a qualified seal system

3. test connections, vents, and drains between containment isolation valves that:

    A. are 1 inch or less in size

    B. administratively secured closed

    C. consist of a double barrier (e.g., two valves in series, one valve with a nipple and cap, one valve and a blind flange)

This guidance may be applied to either Option A or Option B of Appendix J.

For Case No. 2, a qualified seal system is defined in ANSI/ANS-56.8-1994 as a system that is capable of sealing the leakage with a liquid at a pressure no less than 1.1 Pa, (calculated containment peak accident pressure), for at least 30 days following the DB LOCA. The staff's position is the analysis of the sealing capability includes the assumption of the most limiting single failure of any active component. Also, unless there is a virtually unlimited supply of sealing liquid (such as from a suppression pool or recirculation sump), limits for liquid leakage rate should be assigned to these valves based on analysis and included in the plant TS. Periodic leakage rate testing, using the sealing liquid as the test medium, is then needed to ensure that the limits are maintained.

For Case No. 3, to ensure that containment integrity is restored after testing, the test, vent, and drain connections that are used to facilitate local leakage rate testing and the performance of the CILRT should be under administrative control and should be subject to periodic surveillance, to ensure their integrity and to verify the effectiveness of administrative controls.

<u>Operational Programs</u>. For COL reviews, the description of the operational program and proposed implementation milestones for the Containment Leak Rate Testing Program are

reviewed in accordance with 10 CFR Part 50, Appendix J. The implementation milestones are as follows:

A. Appendix J, Option A, Section III:

   1. Type A, B and C test: before any reactor operating period.

B. Appendix J, Option B, Section III.A:

   1. Type A test: after the containment has been completed and is ready for operation.

   2. Type B and C test: before initial criticality.

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 52 requires the reactor containment and associated equipment be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure. GDC 52 applies to DSRS Section 6.2.6 because the review focuses on containment leakage rate testing, which includes the integrated leakage rate testing specified in GDC 52. The requirement for integrated leakage rate testing of the reactor containment is imposed to ensure that it will function as designed in the event of an accident.

   Meeting the requirements of GDC 52 provides assurance that the reactor containment will function as designed and that releases of fission products to the environment will not result in offsite radiation doses in excess of the reference values specified in 10 CFR Part 100, “Reactor Site Criteria.’

2. Compliance with GDC 53 requires the reactor containment be designed to permit (1) periodic inspection of penetrations, (2) an appropriate surveillance program, and (3) periodic testing of the leak tightness of penetrations with resilient seals and expansion bellows.

   GDC 53 applies to this DSRS section because the review broadly covers containment testing. The requirement for inspection, surveillance, and periodic testing of reactor containment penetrations, particularly those with resilient seals and expansion bellows, is imposed because these penetrations are among the containment vessel components most likely to be the source of leakage.

   Meeting the requirements of GDC 53 provides assurance that containment penetrations will function as designed in terms of leakage and will not contribute unduly to offsite radiation doses.

3. Compliance with GDC 54 requires piping systems penetrating primary reactor containment be provided with leak detection, isolation, and performance testing capabilities.

GDC 54 applies to this DSRS section because the review broadly covers containment testing. The requirements of GDC 54 are imposed so unanticipated leakage from piping systems penetrating the reactor containment will not occur during the recovery period that follows a LOCA. Such leakage would compromise the ability of the system to limit the release of fission products to the environment.

Meeting this requirement provides assurance that piping systems penetrating the reactor containment will not be an additional source of leaking fission products and, hence, that releases of fission products off site will not result in radiation doses in excess of the reference values specified in 10 CFR Part 100.

4. Appendix J to 10 CFR Part 50 specifies requirements and acceptance criteria for preoperational and periodic testing of the leak tightness of the reactor containment and penetrations.

   Appendix J applies to this DSRS section because it contains detailed requirements concerning the manner in which the reactor containment and its parts must be tested. These tests include (1) periodic CILRTs, (2) local testing of containment penetration leakage rates, and (3) local testing of isolation valve leakage rates. Appendix J includes pertinent information on the frequency of testing, pressures at which tests will be conducted, recording of test results, and acceptance criteria for testing.

   Meeting the requirements of Appendix J to 10 CFR Part 50 provides assurance that the leak tightness of the containment will be within the values specified in the facility TS and that offsite radiation doses in excess of the reference valves specified in 10 CFR Part 100 will not occur.

5. Meeting the requirements of GDC 5 as it relates to providing assurance that sharing of structures, systems, and components important to safety among nuclear power units will not significantly impair their ability to perform their safety functions.

6. The regulations in 10 CFR 100.11 specify the manner in which exclusion area distance, low population zone distance, and population center distance are determined for a proposed nuclear plant site.

   The containment leakage rate is one of the factors considered when calculating radiation doses associated with accidents. Radiation doses thus calculated determine the acceptability of the exclusion area distance, low population zone distance, and population center distance.

   Verifying the containment leakage rate by means of periodic testing provides assurance that the leakage rate will remain below values assumed in the accident analysis conducted to determine the acceptability of the nuclear power plant site and that offsite radiation doses will be within the reference values specified in 10 CFR Part 100.

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.

At the DC stage, the preliminary design provisions that will permit containment leakage rate testing to be done in accordance with the requirements of Appendix J are reviewed. In some instances, however, the applicant may not be able to address specific aspects of the leakage rate testing program because of incomplete designs. Under these circumstances, the design criteria, and other commitments, that will ensure compliance with the requirements of Appendix J are reviewed. In addition, the applicant's rationale for concluding the requirements of Appendix J will be met is reviewed. If, on the other hand, the applicant is able to address specific aspects of the leakage rate testing program, the review will be akin to the COL stage review, described below, as much as practical considering the detail provided by the applicant.

At the COL stage, the containment final design is reviewed, and it is verified the containment leakage rate testing program meets the requirements of Appendix J. In addition, the plant TS are reviewed for completeness and for conformance to Appendix J.

The review of the reactor containment leakage rate test program at the COL stage specifically includes the following:

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: Light-Water Small Modular 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 small modular reactor (SMR) framework. Using the graded approach described in NUREG-0800, Intro Part 2, the NRC staff may determine that, for certain 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 completing the applicable ITAAC. The use of the selected programs to augment or replace traditional review procedures is shown in Figure 1 of NUREG-0800, Intro Part 2. Examples of such programs that may be relevant to the graded approach for these SSCs include:

    - 10 CFR Part 50, Appendix A, GDC, Overall Requirements, Criteria 1–5
    - 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 Nonsafety 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 DC or COL applications submitted under 10 CFR 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, "Resolution of Generic Safety Issues," 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. <u>CILRT (Type A Test)</u>. Those systems not vented or drained should be identified and the reason for not venting or draining should be stated. System description and piping and instrumentation diagrams are used by the reviewer to confirm, in the vented and drained condition, the isolation valves are exposed to the test air pressure and differential pressure, i.e., the systems are vented and drained both upstream and downstream of the containment isolation valves. For Option B, guidance on venting and draining is available in Section 8.0 of NEI 94-01, Rev. 0.

4. <u>Containment Penetration Leakage Rate Test (Type B Test)</u>. All containment penetrations should be listed in the test program. By reference to piping and instrumentation diagrams, the reviewer confirms all penetrations have been listed. The program should identify any penetration not requiring leakage rate testing and the reason for not requiring a test should be stated. The reviewer confirms those penetrations not requiring testing cannot result in leakage to the atmosphere during a LOCA.

   Test pressures for containment penetrations should be stated in the test program and in the TS. The test pressure is acceptable if it is the calculated peak containment internal pressure related to the DB LOCA.

5. <u>Containment Isolation Valve Leakage Rate Test (Type C Test)</u>. All containment isolation valves requiring a Type C test should be listed in the test program. By reference to the system description schematics, and piping and instrumentation diagrams, the reviewer confirms all isolation valves to be tested have been listed.

   Test pressures for isolation valve Type C tests should be included in the test program and TS.

   Special testing procedures for NuScale SMR containments should be identified. The reviewer confirms the NuScale SMR containment design does not take credit for holdup, dilution or filtering of primary containment leakage.

   The reviewer ensures that the applicant has provided a leakage rate testing program and has specified the maximum leakage rate that may occur from bypass leakage for NuScale SMR containments. Potential leakage paths that bypass the containment or may leak directly to atmosphere must be identified.

   The total amount of containment bypass leakage to the environment must be specified and included in the TS. The reviewer determines the test provisions are adequate to confirm the bypass leakage rate specified.

   Preoperational and periodic tests are reviewed by the appropriate NRC Regional Office.

   In SECY-93-087 the staff recommended the interval for Type C testing be changed from 24 months, as specified in Appendix J, Option A, to 30 months. The Commission approved this recommendation in its staff requirements memorandum (SRM) dated July 21, 1993. Since no applicable revision to Option A has been issued for this position, a partial exemption from Appendix J, Option A, would be required for an applicant to utilize a 30-month interval for Type C testing. Under Option B of Appendix J, RG 1.163 allows Type B and Type C test intervals of 30 months or, under certain circumstances, more.

6. Operational Programs. The reviewer verifies the Containment Leak Rate Testing is fully described, and implementation milestones have been identified. The reviewer verifies the program and implementation milestones are included in the DCD Table 13.x.

   Implementation of this program will be inspected in accordance with NRC Inspection Manual Chapter (IMC)-2504, "Construction Inspection Program—Non-ITAAC Inspections."

7. For review of a DC application, the reviewer should follow the above procedures to verify the design, including requirements and restrictions (e.g., interface requirements and site parameters), set forth in the DCD, meets the acceptance criteria. 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 DCD.

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

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

IV. EVALUATION FINDINGS

The reviewer verifies the applicant has provided sufficient information and 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.

The staff concludes the containment leakage rate testing program is acceptable and meets the requirements of GDCs 5, 52, 53, and 54; Appendix J to 10 CFR Part 50; and 10 CFR Part 100. This conclusion is based on the following:

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 the regulation being discussed):

   A. Meeting the regulatory positions in RG(s);

   B. Providing and meeting an alternative method to regulatory positions in RG, that the staff has reviewed and found to be acceptable;

   C. Using calculational methods for (state what was evaluated) that have been previously 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

   D. Meeting the provisions of (industry standard number and title) that have 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 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.

3. The applicant described the Containment Leak Rate Testing and its implementation in conformance with 10 CFR Part 50, Appendix J.

4. 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.

5. In addition, to the extent 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 SRP revision in effect 6 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 SMR designs, however, differ significantly from large light-water nuclear 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 Staff Requirements Memorandum (SRM)-COMGBJ-10-0004/COMGEA-10-0001, "Use of Risk Insights To Enhance Safety Focus of Small Modular Reactor Reviews," dated August 31, 2010. 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 preapplication activities. Accordingly, the staff has developed the content of the DSRS as an alternative method for evaluating 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 to address new design or siting assumptions.

VI. REFERENCES

1. *U.S. Code of Federal Regulations*, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," Part 50, Title 10, "Energy," Appendix J.

2. 10 CFR Part 50, Appendix A, GDC 5, "Sharing of Structures, Systems and Components."

3. 10 CFR Part 50, Appendix A, GDC 52, "Capability for Containment Leakage Rate Testing."

4. 10 CFR Part 50, Appendix A, GDC 53, "Provisions for Containment Testing and Inspection."

5. 10 CFR Part 50, Appendix A, GDC 54, "Piping Systems Penetrating Containment."

6. *U.S. Code of Federal Regulations*, "Early Site Permits; Standard Design Certifications and Combined Licenses for Nuclear Power Plants," Part 52, Title 10, "Energy."

7. *U.S. Code of Federal Regulations*, "Reactor Site Criteria," Part 100, Title 10, "Energy."

8. U.S. Nuclear Regulatory Commission, "Performance-Based Containment Leak-Test Program," Regulatory Guide (RG) 1.163, September 1995, Agencywide Documents Access and Management System (ADAMS) Accession No. ML003740058.

9. Nuclear Energy Institute, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," NEI 94-01, Revision 3-A, December 2012.

10. American National Standards Institute/American Nuclear Society, "Containment System Leakage Testing Requirements," ANSI/ANS-56.8-1994, August 4, 1994, ADAMS Accession No. ML11327A024.

11. U.S. Nuclear Regulatory Commission, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," SECY 93-087, April 2, 1993, and corresponding SRM dated July 21, 1993. ADAMS Accession Nos. ML003708021 and ML003708056.

12. U.S. Nuclear Regulatory Commission, "Construction Inspection Program—Non-ITAAC Inspections," NRC Inspection Manual Chapter IMC-2504, April 25, 2006, ADAMS Accession No. ML060460204.

13. U.S. Nuclear Regulatory Commission, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," RG 1.160, Revision 3, May 2012, ADAMS Accession No. ML113610098.

14. U.S. Nuclear Regulatory Commission, "Combined License Applications for Nuclear Power Plants (LWR Edition)," RG 1.206.

15. U.S. Nuclear Regulatory Commission, "Guidance for ITAAC Closure under 10 CFR Part 52," RG 1.215, Revision 2, July 2015, ADAMS Accession No. ML15105A447.

16. U.S. Nuclear Regulatory Commission, "Initial Test Programs for Water-Cooled Nuclear Power Plants," RG 1.68, Revision 4, June 2013, ADAMS Accession No. ML13051A027.