U.S. NUCLEAR REGULATORY COMMISSION

DESIGN-SPECIFIC REVIEW STANDARD for NuScale SMR DESIGN

6.2.1.1.A CONTAINMENT

REVIEW RESPONSIBILITIES

Primary - Organization responsible for the review of containment integrity

Secondary - None

I. AREAS OF REVIEW

NuScale is an integral pressurized-water, small modular reactor (SMR) with the reactor, helical coil 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 a smaller reactor vessel. Each containment module is located in the reactor building in its individual bay and is immersed in a large, borated pool of water that serves as the passive containment heat sink and ultimate heat sink.

The specific areas of review are as follows:

1. The temperature and pressure conditions in the containment due to a spectrum (including break size and location) of postulated loss-of-coolant accidents (LOCAs) (i.e., reactor coolant system pipe breaks) and secondary system steam and feedwater line breaks.

2. The maximum expected external pressure to which the containment may be subjected.

3. The effects of minimum containment pressure and the timing of pressure changes that are used in analyses of emergency core cooling system (ECCS) capability.

4. The effectiveness of static (passive) and active heat removal mechanisms.

5. The range and accuracy of instrumentation that is provided to monitor and record containment conditions during and following an accident.

6. Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC). For design certification (DC) and combined license (COL) reviews, the staff reviews the applicant's proposed ITAAC associated with the structures, systems, and components (SSCs) related to this 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 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

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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.

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

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

Review Interfaces

Other DSRS sections interface with this section as follows:

1. The risk to containment integrity from explosions or releases from nearby chemical plants using reactor cogenerated steam for process heat under SRP Sections 2.2.1-2.2.2, 2.2.3, 3.5.2, and 19.0.

2. The electrical design of the instrumentation provided to monitor and record containment conditions during and after an accident, and the effectiveness of the administrative controls and the instrumentation and control provisions to prevent inadvertent operation of the containment heat removal systems or system trains under DSRS Sections 7.0, 7.1, and 7.2.

3. The design adequacy of the containment vessel and its internal structures under DSRS Section 3.8.2.

4. The design adequacy of mechanical components and their supports under SRP Section 3.9.3.

5. The proposed technical specifications that pertain to the surveillance requirements for spring- or weight-loaded check valves used in an evacuated containment and vacuum isolation valves under DSRS Section 16.0.

6. The environmental qualification (EQ) of the containment system under DSRS Section 3.11.

7. Offsite and control room dose under DSRS Section 15.0.3.

8. For new plant applicants, shutdown risk assessment reviews, including containment analysis issues, under SRP Sections 19.0 and 19.3.

9. The effects of static and dynamic hydraulic forces on the reactor building pool, the reactor module bays, and the enclosed containment vessel caused by tsunami hazards is performed under SRP Section 2.4.6.

10. The effects of groundwater on the underground reactor building pool, the reactor module bays, and the enclosed containment structure vessel, including effects of groundwater levels, piezometric-hydraulic heads and other hydraulic effects of groundwater on the design bases of subsurface safety-related and risk-significant SSCs, are performed under SRP Section 2.4.12.

II. ACCEPTANCE CRITERIA

Requirements

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

1. General Design Criterion (GDC) 4, as it relates to SSCs important to safety to be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including LOCAs.

2. GDC 5, as it relates to SSCs important to safety, which shall not be shared among nuclear power units or modules in a single power unit unless it can be shown that such sharing will not significantly impair their ability to perform their safety or risk-significant functions, including, in the event of an accident in one unit or module, an orderly shutdown and cooldown of the remaining units or modules.

3. GDC 16, as it relates to the reactor containment and associated systems being designed to assure that containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

4. GDC 50, as it relates to the reactor containment structure and associated heat removal system(s) being designed so that the containment structure and its internal compartments can accommodate the calculated pressure and temperature conditions resulting from any LOCA without exceeding the design leakage rate and with sufficient margin.

5. GDC 38, as it relates to the containment heat removal system(s) function to rapidly reduce the containment pressure and temperature following any LOCA and maintain them at acceptably low levels.

6. GDC 13, as it relates to instrumentation and control to require instrumentation be provided to monitor variables and systems over their anticipated ranges for normal operation and for accident conditions as appropriate to assure adequate safety.

7. GDC 64, as it relates to monitoring radioactivity releases to require means be provided for monitoring the reactor containment atmosphere for radioactivity that may be released from normal operations and from postulated accidents.

8. Title 10 of the *Code of Federal Regulations* (10 CFR), 50.34, “Contents of Applications; Technical Information”, paragraph (f)(3)(v)(A)(1), “Additional TMI Related Requirements,” as it relates to containment integrity being maintained during an accident that releases hydrogen generated from a 100-percent fuel-clad metal-water reaction accompanied by hydrogen burning.

9. 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 plant that incorporates the DC is built and will operate in accordance with the DC, the provisions of the Atomic Energy Act, and the U.S. Nuclear Regulatory Commission’s (NRC’s) regulations.

10. 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 Atomic Energy Act, 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. To satisfy the requirements of GDC 16 and 50 regarding sufficient design margin, for plants in the design stage (i.e., at the construction permit (CP) or design certification (DC) stage) of review, the containment design pressure should provide at least a 10-percent margin above the accepted peak calculated containment pressure following a LOCA, or a steam or feedwater line break. Design margins of less than 10 percent may be sufficient, provided appropriate justification is provided. For plants at the operating license (OL) or COL stage of review, the peak calculated containment pressure following a LOCA, or a steam or feedwater line break, should be less than the containment design pressure.

2. To satisfy the requirements of GDC 38 to rapidly reduce the containment pressure, the containment pressure should be reduced to less than 50 percent of the peak calculated pressure for the design-basis LOCA within 24 hours after the postulated accident. If analysis shows that the calculated containment pressure may not be reduced to 50 percent of the peak calculated pressure within 24 hours, the organization responsible for the review of DSRS Section 15.0.3 should be notified.

3. To satisfy the requirements of GDC 38 and 50 with respect to the containment heat removal capability and design margin, the LOCA analysis should be based on the assumption of loss of offsite power and the most severe single failure in the emergency power system (e.g., a diesel generator failure), the containment heat removal systems (e.g., a fan, pump, or valve failure), or the core cooling systems (e.g., a pump or valve failure). The selection made should result in the highest calculated containment pressure.

4. To satisfy the requirements of GDC 38 and 50 with respect to the containment heat removal capability and design margin, the containment response analysis for postulated secondary system pipe ruptures should be based on the most severe single failure of the secondary system isolation provisions (e.g., main steam isolation valve failure or feedwater line isolation valve failure). The analysis should also be based on a spectrum of pipe break sizes and reactor power levels. The accident conditions selected should result in the highest calculated containment pressure or temperature depending on the purpose of the analysis. Acceptable methods for the calculation of the containment environmental response to main steam line break accidents are found in NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment."

5. To satisfy the requirements of GDC 38 and 50 with respect to the functional capability of the containment heat removal systems and containment structure under LOCA conditions, provisions should be made to protect the containment structure against possible damage from external pressure conditions that may result. The provisions made should include conservative structural design to assure that the containment structure is capable of withstanding the maximum expected external pressure, or interlocks in the plant protection system and administrative controls to preclude inadvertent operation of the systems. If the containment is designed to withstand the maximum expected external pressure, the external design pressure of the containment should provide an adequate margin above the maximum expected external pressure to account for uncertainties in the analysis of the postulated event.

6. In accordance with the requirements of GDC 13 and 64, and 10 CFR 50.34(f)(2)(xvii), instrumentation capable of operating in the postaccident environment should be provided to monitor the containment atmosphere pressure and temperature, and the sump water level and temperature, following an accident. The instrumentation should have adequate range, accuracy, and response to assure that the above parameters can be tracked and recorded throughout the course of an accident. See Item II.F.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," and NUREG-0718, "Licensing Requirements for Pending Applications for Construction Permits and Manufacturing License Applications," and Branch Technical Position 7-10, "Guidance on Application of Regulatory Guide 1.97."

7. In accordance with GDC 4, containment internal structures and system components (e.g., integral reactor vessel with pressurizer, steam generators) and supports should be designed to withstand the differential pressure loadings that may be imposed as a result of pipe breaks within the containment. In addition, the containment vessels submerged in a common reactor building pool that is below grade should be able to accommodate

hazards arising from floods, tsunamis, and accidents generating missiles, and chemical releases from nearby industrial facilities.

8. In meeting the requirements of 10 CFR 50.34(f)(3)(v)(A)(1), applicants subject to this section should evaluate an accident that releases hydrogen generated from a 100-percent fuel clad metal-water reaction. The evaluation should demonstrate that the appropriate article for Service Level C Limits (considering pressure and dead load only) for steel containments, from American Society of Mechanical Engineers (ASME) Boiler Pressure Vessel Code, Section III, are met. In addition to the containment pressurization caused directly by this accident, the increase in pressure from hydrogen burning in containment should be analyzed.

9. To satisfy the requirements of GDC 5, an accident that affects one containment vessel in the set of reactor modules should not impair the containment integrity of any other reactor module in the common reactor building pool. In the event of a prolonged station blackout (SBO), the reactor building pool should have a sufficient capacity to accommodate cooling of all reactor modules the building pool contains, assuming simultaneous shutdown of all modules from full power.

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 16 requires the containment be designed as a leaktight barrier that will withstand the most extreme accident conditions for the duration of any postulated accident. This DSRS section evaluates the peak pressure and temperature conditions for which the containment should be designed. The containment should be leaktight and able to withstand accidents because it is the final barrier against the release of radioactivity to the environment. Because the primary reactor containment vessel is the final barrier of the defense-in-depth concept to protect against the uncontrolled release of radioactivity to the environment, preserving containment integrity under the dynamic conditions imposed by postulated LOCAs is essential. Meeting GDC 16 provides assurance that radioactivity will not be released to the environment.

2. GDC 50 requires the containment structure and associated heat removal system be designed with margin to accommodate any LOCA such that the containment design leak rate is not exceeded. A LOCA potentially causes the greatest pressure surge and release of fission products when compared to any other accident. Because it is the most severe challenge expected, containment should be designed to definitively withstand this accident. Meeting GDC 50 will ensure that containment integrity is maintained under the most severe accident conditions, thus precluding the release of radioactivity to the environment.

3. GDC 38 requires the establishment of a containment heat removal system that will rapidly reduce containment pressure and temperature following any LOCA. The containment heat removal system supports the containment function by minimizing the duration and intensity of the pressure and temperature increase following a LOCA, thus

lessening the challenge to containment integrity. Meeting GDC 38 will help ensure that the containment can fulfill its role as the final barrier against the release of radioactivity to the environment. Because there is a common passive containment heat removal system for multiple reactor modules, each in a separate containment vessel, the common containment heat removal system should also meet GDC 5 in terms of capacity and duration of functional capability.

4. GDC 13 requires that instrumentation be provided to monitor all expected parameters of normal operation, anticipated operational occurrences, and accidents to assure adequate reactor safety is maintained. Because containment plays a vital safety role, appropriate instrumentation, such as temperature and pressure, should be provided so that operators can verify containment is properly fulfilling its function. Regulatory Guide (RG) 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," provides specific criteria for the design of containment instrumentation which have been found acceptable by the NRC as fulfilling GDC 13. Meeting GDC 13 and the specific guidance of RG 1.97 will help ensure that containment accomplishes its mission of precluding the release of radioactivity to the environment. DSRS Section 7.2 provides the specific acceptance criteria to satisfy RG 1.97.

5. GDC 64 requires that the containment atmosphere be monitored for the release of radioactivity from normal operations, anticipated operational occurrences, and accidents. In order to ensure that the containment functions properly, operators should be aware of any radioactive releases within containment so that they can take appropriate manual action or monitor the automatic isolation action. RG 1.97 provides specific criteria for the design of containment instrumentation which have been found acceptable by the NRC as fulfilling GDC 64. DSRS Section 7.2 provides the specific acceptance criteria to satisfy RG 1.97. Meeting GDC 64 and the specific guidance of RG 1.97 will assist operators in ensuring that containment meets its safety function of preventing the release of radioactivity to the environment.

6. 10 CFR 50.34(f)(3)(v)(A)(1) requires that the containment be designed to withstand hydrogen burning during an accident that releases hydrogen from a 100-percent fuel clad metal-water reaction. During the accident at Three Mile Island, Unit 2, metal-water reactions generated hydrogen in excess of the amounts originally anticipated. As a result of this finding, the Commission issued requirements on hydrogen control in 10 CFR 50.34(f). Other criteria require the containment to be designed to withstand postulated accidents. If such a postulated accident releases or generates hydrogen, an added containment pressurization effect beyond the initial accident may be experienced due to the release of hydrogen and in-leakage oxygen or radiolysis of water. The containment should be designed to withstand this additional pressure to ensure that its integrity is maintained, thus precluding the release of radioactivity to the environment.

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: 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 SSCs 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 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, General Design Criteria (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. Upon request from the primary reviewer, the reviewer of an interfacing DSRS section may provide input to address an area of review stated in Subsection I of this DSRS section. The primary reviewer obtains and uses such input as required to assure that this review is complete.

4. The primary review organization reviews the containment response analyses to determine the acceptability of the calculated containment design pressure and temperature, and in addition, the containment depressurization time. The organization responsible for the review of DSRS Section 15.0.3 should be notified if the containment depressurization time does not meet the acceptance criterion. The primary review organization for this DSRS section reviews the assumptions made in the analyses to maximize the calculated containment pressure and temperature. The primary review organization for this DSRS section determines the conservatism of the respective containment response analyses by comparing the analytical models, and the assumptions made, with the acceptance criteria in Subsection II of this DSRS section and by performing appropriate confirmatory analyses. It is not necessary to perform independent accident pressure calculations. The primary review organization for this DSRS section will ascertain, however, whether the adequacy of the applicant's calculation model has been demonstrated. The primary review organization for this DSRS section determines whether the applicant has identified the pipe break(s) resulting in the highest containment pressure and temperature. Breaks of the reactor coolant system and secondary system steam and feedwater line breaks should be analyzed by the applicant. The primary review organization for this DSRS section reviews the assumptions used to determine whether the analyses are acceptably conservative. Design certification applicants should meet the CP containment design pressure margin criterion of Item 1 of the DSRS Acceptance Criteria.

5. The primary review organization verifies that the containment is designed to preclude hydrogen burning during an accident that releases hydrogen from a 100-percent fuel clad metal-water reaction as described in Item 8 of the DSRS Acceptance Criteria.

6. The primary review organization performs confirmatory containment response analyses when necessary. The purpose of these analyses is to confirm the applicant's predictions of the response of the containment to LOCAs and main steam and feedwater line breaks. In general, the primary review organization analyzes only the limiting primary leak size from the integral reactor vessel and limiting feedwater or steam line pipe breaks (i.e., the leak size or pipe breaks that establish the containment design pressure and containment depressurization time). However, if in the reviewer's judgment the worst leak size or break has not been identified, other leaks or pipe breaks will be analyzed.

7. The primary review organization reviews analyses of the external pressure of the containment structure caused by pressure and temperature changes inside the containment due to the continuous operation of containment heat removal systems during an accident. The primary review organization determines whether the most severe condition has been identified and whether the analysis was done in a conservative manner. For plants at the DC stage of review, the external design pressure margin should be at least 10 percent. For plants at the COL stage of review, the maximum expected external pressure should be less than the containment external design pressure. In general, the maximum expected external pressure should be approximately the same as at the CP or DC stage of review. However, revised or upgraded analytical models or minor changes in the as-built design of the plant may result in a decrease in the margin. If the primary containment is not designed to withstand the maximum external pressure, the primary review organization will evaluate the acceptability of the provisions made in the plant design to mitigate or withstand the consequences of the above postulated events, and will evaluate, in conjunction with the primary reviewer for DSRS Section 7.2, the administrative controls and instrumentation and control provisions to preclude these events.

8. The primary review organization for this DSRS section reviews the accuracy and range of the instrumentation provided to monitor the postaccident environment. The primary review organization for DSRS Section 7.2 and the primary review organization for DSRS Section 3.11 have review responsibility for the acceptability of, and the qualification test program for the sensing and actuation instrumentation of, the plant protection system and the postaccident monitoring instrumentation and recording equipment.

9. The containment analyses should also consider shutdown conditions, when appropriate, to ensure that a basis is provided for procedures, instrumentation, operator response, equipment interactions, and equipment response during shutdown operations. The analyses should encompass shutdown thermodynamic states and physical configurations to which the plant can be subjected during shutdown conditions (such as containment closure time, temperature and time to uncover the core during loss of decay heat removal). Because shutdown for refueling operations involves filling the containment vessel with borated reactor building pool water, the application should address removal of the borated water and the steps needed to clean the inside of the containment vessel and the reactor vessel outer wall to prevent boric acid corrosion or crystal formation.

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

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

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

IV. EVALUATION FINDINGS

The reviewer verifies that the applicant has provided sufficient information and that the review and calculations (if applicable), 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 SER. The reviewer also states the bases for those conclusions.

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 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 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 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 in order to address new design or siting assumptions.

VI. REFERENCES

1. *U.S. Code of Federal Regulations*, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," Section 46, Part 50, Chapter I, Title 10, "Energy," and 10 CFR Part 50, Appendix K, "ECCS Evaluation Models."

2. 10 CFR Part 50, Appendix A, General Design Criterion 4, "Environmental and Dynamic Effects Design Bases."

3. 10 CFR Part 50, Appendix A, General Design Criterion 5, "Sharing of Structures, Systems, and Components."

4. 10 CFR Part 50, Appendix A, General Design Criterion 13, "Instrumentation and Control."

5. 10 CFR Part 50, Appendix A, General Design Criterion 16, "Containment Design."

6. 10 CFR Part 50, Appendix A, General Design Criterion 38, "Containment Heat Removal."

7. 10 CFR Part 50, Appendix A, General Design Criterion 39, "Inspection of Containment Heat Removal System."

8. 10 CFR Part 50, Appendix A, General Design Criterion 40, "Testing of Containment Heat Removal System."

9. 10 CFR Part 50, Appendix A, General Design Criterion 50, "Containment Design Basis."

10. 10 CFR Part 50, Appendix A, General Design Criterion 64, "Monitoring Radioactivity Releases."

11. U.S. Nuclear Regulatory Commission, "Combined License Applications for Nuclear Power Plants (LWR Edition)," Regulatory Guide (RG) 1.206, (ADAMS Accession No. ML070720184).

12. U.S. Nuclear Regulatory Commission, "Guidance for ITAAC Closure under 10 CFR Part 52," RG 1.215 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML112580018).

13. "RELAP4 MOD5, A Computer Program for Transient Thermal Hydraulic Analysis of Nuclear Reactors and Related Systems User's Manual," ANCR-NUREG-1335, September 1976.

14. U.S. Nuclear Regulatory Commission, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," NUREG-0588, Rev. 1, July 1981.

15. U.S. Nuclear Regulatory Commission, "Asymmetric Blowdown Loads on PWR Primary Systems," NUREG-0609, January 1981.

16. U.S. Nuclear Regulatory Commission, "COMPARE: A Computer Program for the Transient Calculation of a System of Volumes Connected by Flowing Vents," LA-NUREG-6488-MS, September 1976.

17. U.S. Nuclear Regulatory Commission, "Licensing Requirements for Pending Applications for Construction Permits and Manufacturing License," NUREG-0718, Rev. 1, March 1981.

18. U.S. Nuclear Regulatory Commission, "Clarification of TMI Action Plan Requirements," NUREG-0737, November 1980.

19. Moody, F.J., "Maximum Flow Rate of a Single Component, Two-Phase Mixture," *Journal of Heat Transfer*, 87(1):134–141, February 1965.

20. U.S. Nuclear Regulatory Commission, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," RG 1.97, (ADAMS Accession No. ML061580448).

21. U.S. Nuclear Regulatory Commission, "Minimum Containment Pressure Model for PWR ECCS Performance Evaluation." Branch Technical Position 6-2 (ADAMS Accession No. ML070740442).

22. American Society of Mechanical Engineers, *Boiler and Pressure Vessel Code*, Section III, Division 1, Subsection NE, “Class MC Components.”

23. Uchida, H., A. Oyama, and Y. Togo, “Evaluation of Post-Incident Cooling Systems of Light-Water Power Reactors,” *Proceedings of the Third International Conference on the Peaceful Uses of Atomic Energy*, Volume 13, Session 3.9, United Nations, Geneva, 1964.

24. Hedrick, R.A., Cudlin, J.J., Holtz, R.C., Babcock and Wilcox Company, “CRAFT 2: Fortran Program for Digital Simulation of a Multinode Reactor Plant during Loss-of-Coolant,” BAW 10092, December 1974.

25. U.S. Nuclear Regulatory Commission, “Code Manual for CONTAIN 2.0: A Computer Code for Nuclear Reactor Containment Analysis,” NUREG/CR-6533, December 1997.

26. Numerical Applications, Inc., “GOTHIC: Containment Analysis Package User Manual, Qualification Report and Technical Manual,” NAI 8907.

27. McMurtray, Anthony C., U.S. Nuclear Regulatory Commission, letter to Thomas Coutu, Nuclear Management Company, LLC, September 29, 2003 (ADAMS Accession No. ML032681050).

28. U.S. Nuclear Regulatory Commission, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” RG 1.183 (ADAMS Accession No. ML003716792).

29. U.S. Nuclear Regulatory Commission, “Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors,” RG 1.195 (ADAMS Accession No. ML031490640).

30. U.S. Nuclear Regulatory Commission, “Loss of Decay Heat Removal,” Generic Letter 88-17, October 17, 1988.