Draft for Comment

U.S. NUCLEAR REGULATORY COMMISSION

DESIGN-SPECIFIC REVIEW STANDARD FOR NuScale SMR DESIGN

6.2.1.3 MASS AND ENERGY RELEASE ANALYSIS FOR POSTULATED LOSS-OF-COOLANT ACCIDENTS (LOCAs)

REVIEW RESPONSIBILITIES

Primary - Organization responsible for the review of containment integrity

Secondary - None

I. AREAS OF REVIEW

The NuScale design 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 reactor pressure vessel (RPV). The NuScale reactor containment is a low alloy steel evacuated vessel that surrounds the RPV and is immersed in a large borated-water reactor building pool that serves as the Ultimate Heat Sink for the passive Emergency Core Cooling System (ECCS), the passive Containment Heat Removal System, and the passive Decay Heat Removal System (DHRS). The DHRS has heat exchangers located in the reactor building pool, which houses the containment vessel. The pool-immersed containment vessel for each reactor module is housed in a three-sided bay that opens on one side into the main portion of the reactor building pool.

There is no large diameter piping in the reactor coolant pressure boundary (RCPB). The small diameter piping in the RCPB includes the inlet and outlet piping for the Chemical and Volume Control System (CVCS), the injection piping for the Shutdown Accumulator System (SAS), and the reactor pressure vessel vent for non-condensable gas being expelled during startup. In addition, the RCPB includes redundant/duplicate valves for the ASME Code safety relief valves (SRVs), the reactor vent valves (RVVs), which allow controlled depressurization of the reactor vessel to the containment vessel, and the reactor recirculation valves (RRVs) located near the bottom of the integral reactor vessel. The RRVs are located above the core to prevent draining the RPV below the reactor core and allow condensed water in the containment vessel sump to recirculate into the core.

Any break or leak from the RCPB will be a small break loss of coolant accident (SBLOCA). If the SBLOCA is in the upper portion of the reactor vessel and is sufficiently small to preclude the voiding of reactor coolant below the primary system inlet to the steam generators, the DHRS heat exchangers will provide cooling to the core to effect cooldown and depressurization. For any other RCPB break, the passive cooling provided through the containment vessel wall, which is immersed in the cold reactor building pool, will condense released reactor steam and provide sufficient condensate return to the containment vessel sump for recirculation through the RRVs back into the reactor once pressure is equalized between the reactor vessel and containment vessel. The RVVs provide a means for the actuation system or operators to accelerate reactor vessel depressurization and the equalization of pressure between the reactor vessel and containment vessel. The containment vessel is designed to accommodate the highest possible pressure from reactor vessel depressurization.

The analyses of the mass and energy release are reviewed to assure that the data used to evaluate the containment functional design are acceptable for that purpose.

The specific areas of review are as follows:

1. The energy sources that are available for release to the containment.

2. The mass and energy release rate calculations for the initial blowdown phase of the accident.

3. Because of the additional steam generator stored energy available for release, the mass and energy release rate calculations for the reactor depressurization and core reflood and post-reflood phases of the accident.

4. 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 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.

5. 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. Review of the acceptability of piping design criteria and selected break locations and break sizes based on the provisions made to limit pipe motion for breaks postulated to occur, is performed under SRP Section 3.6.2, including breaks postulated to occur in the short distance between the reactor vessel nozzle and the containment vessel penetration fixture.

2. Determination of SSC risk significance is performed under SRP Sections 19.0 and 19.3.

II. ACCEPTANCE CRITERIA

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

1. General Design Criterion 50, as it relates to the containment being designed with sufficient margin, requires that the containment and its associated systems can accommodate, without exceeding the design leakage rate, and the containment design can withstand the calculated pressure and temperature conditions resulting from any LOCA.

2. 10 CFR Part 50, Appendix K, as it relates to sources of energy during the LOCA, provides requirements to assure that all the energy sources have been considered.

3. 10 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 design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.

4. 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 combined license, 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. General Design Criterion 50 and Appendix K to 10 CFR Part 50

    A. Sources of Energy. The sources of stored and generated energy that should be considered in analyses of LOCAs include: reactor power; decay heat; stored energy in the core; stored energy in the reactor coolant system metal, including the reactor vessel and reactor vessel internals; metal-water reaction energy; and stored energy in the secondary system including the steam generator tubing and secondary water.

    Calculations of the energy available for release from the above sources should be done in general accordance with the requirements of 10 CFR Part 50,

Appendix K, paragraph I.A. However, additional conservatism should be included to maximize the energy release to the containment during the blowdown and reflood phases of a LOCA. An example of this would be accomplished by maximizing the sensible heat stored in the reactor coolant system (RCS) and steam generator metal and increasing the RCS and steam generator secondary mass to account for uncertainties and thermal expansion.

The requirements of paragraph I.B in Appendix K to 10 CFR Part 50, concerning the prediction of fuel clad swelling and rupture should not be considered. This will maximize the energy available for release from the core.

B. Break Size and Location

i. The staff’s review of the applicant's choice of break locations and types is discussed in SRP Section 3.6.2.

ii. Of several breaks postulated, the break selected as the reference case should yield the highest mass and energy release rates, consistent with the criteria for establishing the break location and area.

iii. Containment design basis calculations should be performed for a spectrum of possible pipe break sizes and locations to assure that the worst case has been identified.

C. Calculations. In general, calculations of the mass and energy release rates for a LOCA should be performed in a manner that conservatively establishes the containment internal design pressure (i.e., maximizes the post-accident containment pressure response). The criteria given below for each phase of the accident indicate the conservatism that should exist.

i. Containment Analysis

The analytical approach used to compute the mass and energy release profile will be accepted if both the computer program and volume noding of the reactor, piping and containment systems are similar to those of an approved ECCS analysis. The computer programs that are currently acceptable include CRAFT-2, and RELAP5, when a flow multiplier of 1.0 is used with the applicable choked flow correlation. An alternate approach, which is also acceptable, is to assume a constant blowdown profile using the initial conditions with an acceptable choked flow correlation.

ii. Initial Blowdown Phase Containment Design Basis

The initial mass of water in the reactor coolant system should be based on the reactor coolant system volume calculated for the temperature and pressure conditions assuming that the reactor has been operating continuously at a power level at least 1.02 times the licensed power level (to allow for instrumentation error). An assumed power level lower than the level specified (but not less than the licensed power level) may be used provided the proposed alternative value has been demonstrated to account for uncertainties due to power level instrumentation error.

Mass release rates should be calculated using a model that has been demonstrated to be conservative by comparison to experimental data.

Calculations of heat transfer from surfaces exposed to the primary coolant should be based on nucleate boiling heat transfer. For surfaces exposed to steam, heat transfer calculations should be based on forced convection.

Calculations of heat transfer from the secondary coolant to the steam generator tubes should be based on natural convection heat transfer for tube surfaces immersed in water and condensing heat transfer for the tube surfaces exposed to steam.

Calculations of heat transfer from released reactor steam into the containment wall should be based on forced convection in regions where there is steam jetting and natural convection at points away from jetting. Calculations of heat transfered from condensed reactor water in the containment sump into the containment wall and from the reactor vessel wall into the pooled sump water should be based on natural convection. Heat transfer through the containment vessel wall into the reactor building pool should be based on conduction and natural convection with nucleate boiling.

iii. Core Reflood Phase (Cold Leg RRV Penetration Breaks Only)

Following initial blowdown through a failed RRV, which includes the period from the accident initiation (when the reactor is in a steady-state full power operation condition) to the time that the reactor coolant system equalizes to the containment pressure, the water remaining in the reactor vessel should be assumed to be saturated. Justification should be provided for the refill period, which is the time from the end of the blowdown to the time when flow from the condensed water in the containment vessel sump comes back through the RRVs into the reactor vessel.

Calculations of the core flooding rate should be based on the ECCS operating condition during the core reflood phase, which begins when the water starts to flood the core and continues until the core is completely quenched, or the post-reflood phase, which is the period after the core has been quenched and energy is released to the RCS primary system by the RCS metal, core decay heat, and the steam generators, that maximizes the containment pressure.

Calculations of liquid entrainment, i.e., the carryout rate fraction, which is the mass ratio of liquid exiting the core to the liquid entering the core, should be based on the NuScale full length emergency cooling heat transfer experiments or conservatively scaled-up test results from subscale test.

The assumption of steam quenching should be justified by comparison with applicable experimental data. Liquid entrainment calculations should consider the effect on the carryout rate fraction of the increased core inlet water temperature caused by steam quenching assumed to occur from mixing with the ECCS water.

Steam leaving the steam generators should be assumed to be superheated to the temperature of the secondary coolant.

iv. Post-Reflood Phase

All remaining stored energy in the primary and secondary systems should be removed during the post-reflood phase.

Steam quenching on the containment vessel walls, and due to pressure equalization between the reactor vessel and the containment vessel, should be justified by comparison with applicable experimental data.

The results of post-reflood analytical models should be compared to applicable experimental data.

v. Decay Heat Phase

The dissipation of core decay heat should be considered during this phase of the accident. The fission product decay energy model is acceptable if it is equal to or more conservative than the decay energy model given in DSRS Section 9.2.5.

Steam from decay heat boiling in the core should be assumed to flow to the containment by the path which produces the minimum amount of mixing with the condensed water flowing from the containment sump into the reactor vessel through the RRVs.

Methods for calculating the mass and energy releases for containment design basis calculations should be conservative for these calculations.

2. 10 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 plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.

3. 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 combined license, the provisions of the Atomic Energy Act, and the NRC's regulations. 10 CFR 52.47(a)(1)(vi) provides the requirement for ITAAC for design certification reviews.

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 50 requires the containment structure and associated heat removal system to 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. Since it is the most severe challenge expected, containment must be designed to definitively withstand this accident. Following GDC 50 will ensure that containment integrity is maintained under the most severe accident conditions thus precluding the release of radioactivity to the environment.

2. Appendix K to 10 CFR 50 provides required and acceptable features of evaluation models used to analyze various circumstances applicable to the ECCS. Section I.A of Appendix K provides a comprehensive list of LOCA heat (energy) sources and the reactor operating history assumptions associated with those heat sources. Since the mass and energy release analysis for postulated LOCAs is used to design containment such that it will withstand the worst case LOCA, it is critical that all potential energy sources are taken into account. Following 10 CFR 50 Appendix K will ensure that containment is designed to accommodate all energy sources for the worst case LOCA, thus precluding the potential release of radioactivity to the environment following such a LOCA.

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.

The reviewer confirms, with the lead reviewer for SRP Section 3.6.2, the validity of the applicant's analysis of break/leak size, type and locations for the containment showing the routing of lines containing high energy fluids. The reviewer determines that an appropriate reference case for containment analysis has been identified. In the event a break other than a double-ended pipe rupture is postulated by the applicant, the lead reviewer for SRP Section 3.6.2 will evaluate the applicant’s justification for assuming a leak or limited displacement pipe break.

The reviewer compares the sources of energy considered in the loss-of-coolant analysis and the methods and assumptions used to calculate the energy available for release from the various sources with the acceptance criteria listed in Section II, above. The reviewer determines the acceptability of the analytical models and the assumptions used to calculate the rates of mass and energy release during the initial blowdown, core reflood, and post-reflood phases of a

LOCA. The reviewer also compares energy inventories at various times during a LOCA to ensure that the energy from the various sources has been accounted for and has been transferred to the containment on an appropriate time scale.

The reviewer reviews comparisons made by the applicant to experimental data and makes comparisons to other available experimental data to determine the amount of conservatism in the mass and energy release models.

The reviewer may perform confirmatory analyses of the mass and energy profiles. The purpose of the analysis is to confirm the predictions of the mass and energy release rates appearing in the safety analysis report, and to confirm that an appropriate break location has been considered in these analyses.

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 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 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, DSRS 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.

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.

1. 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.

IV. EVALUATION FINDINGS

The conclusions reached on completion of the review of this DSRS section are presented in DSRS Section 6.2.1.1.

V. IMPLEMENTATION

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

In view of the differences between the designs of SMRs and the designs of large light-water power reactors, the Commission issued SRM- COMGBJ-10-0004/COMGEA-10-0001, "Use of

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

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

VI. REFERENCES

1. Regulatory Guide1.68, "Initial Test Programs for Water-Cooled Nuclear Power Plants."

2. Regulatory Guide1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants."

3. Regulatory Guide1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants."

4. Regulatory Guide1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)."

5. Regulatory Guide1.215, "Guidance for ITAAC Closure Under 10 CFR Part 52."