U.S. NUCLEAR REGULATORY COMMISSION

DESIGN-SPECIFIC REVIEW STANDARD for NuScale SMR DESIGN

15.6.5 LOSS-OF-COOLANT ACCIDENTS RESULTING FROM SPECTRUM OF POSTULATED PIPING BREAKS WITHIN THE REACTOR COOLANT PRESSURE BOUNDARY

REVIEW RESPONSIBILITIES

Primary - Organization responsible for the review of pressurized-water reactor systems

Secondary - Organization responsible for the review of containment

I. AREAS OF REVIEW

The specific areas of review are as follows:

1. Loss-of-coolant accidents (LOCAs) are postulated accidents that would result from the loss of reactor coolant, at a rate in excess of the capability of the normal reactor coolant makeup system, from piping breaks in the reactor coolant pressure boundary. The piping breaks are postulated to occur at various locations and include a spectrum of break sizes, up to a maximum pipe break equivalent in size to the double-ended rupture of the largest pipe in the reactor coolant pressure boundary. Loss of significant quantities of reactor coolant would prevent heat removal from the reactor core, unless the water is replenished. An additional safety concern is the potential for build-up of boric acid because of coolant vaporization in a pressurized-water reactor (PWR). NuScale uses soluble boron as a reactivity control element. If left uncontrolled, the boron concentration could reach precipitation limits and block the coolant channels in the core, preventing heat removal for any size break.

   The NuScale reactor coolant system (RCS) includes a reactor vessel with two integral once-through helical coil steam generators (SGs). The reactor vessel is enclosed in a containment vessel that sits in a pool of surrounding water. Primary system coolant flows by natural convection up through the core toward the top of the vessel, turns downward, and passes through the shell side of the SG before returning to the core. Four feedwater and four steam lines penetrate the vessel providing the secondary flow to the tube-side of the two steam generators to remove the heat generated in the core (two feedwater and two steam lines per generator). An integral pressurizer is at the top of the reactor pressure vessel.

   The NuScale emergency core cooling system (ECCS) recirculates a portion of the reactor coolant inventory to containment to transfer decay heat from the RCS to the reactor pool building, which is the ultimate heat sink. ECCS has two reactor vent valves (RVVs) on the reactor vessel head to transfer decay heat from the RCS to the containment. Additionally, the ECCS has two reactor recirculation valves (RRVs) attached to the lower part of the reactor vessel to allow recirculation of inventory retained in containment to keep the core covered. The ECCS is not designed to add inventory to

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the reactor coolant system (e.g., there are no accumulators or in-containment storage tank).

The containment transfers heat to the ultimate heat sink (UHS), which is the surrounding reactor building pool, using passive heat removal mechanisms (i.e., natural circulation and conduction). Penetrations of the primary system boundary include those associated with the chemical volume and control system (CVCS).

As discussed above, the NuScale ECCS provides core decay heat removal by transferring mass and heat to containment, which eventually transfers heat to the UHS. The ECCS contains the RVVs, which release steam generated in the reactor core from the reactor vessel to the containment where it condenses on the containment's internal surface. The condensate collects in the lower region of the containment vessel. When the liquid level in the containment rises above the top of the recirculation valves, the RRVs provide a natural circulation path from the lower containment through the core and out the RVVs. The steam transfers energy out of the reactor vessel to the containment, then condenses and collects in the bottom of the containment. The RRVs provide for recirculation of water from the containment to the reactor core. The reactor pool serves as the ultimate heat sink. For the first 72 hours after initiation of the accident sequence, only passive, safety-related systems are credited in the design-basis accident (DBA) analysis.

General Design Criterion (GDC) 35, "Emergency Core Cooling," requires each reactor to be equipped with an ECCS while Title 10 of the *Code of Federal Regulations* (10 CFR) 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," provides acceptance criteria for the ECCS. The analysis of ECCS performance has an effect on the design of the piping and support structures for the reactor coolant system, the design of the steam generators and the containment design. The ECCS performance relies on the coupling between reactor coolant system and containment during a LOCA. The result is that stored water might not be needed to flood the reactor pressure vessel to keep the core covered or support recirculation from the containment volume to the reactor pressure vessel to keep the core covered.

The review of the applicant's analysis of the spectrum of postulated LOCAs is closely associated with the review of the ECCS, as described in DSRS Section 6.3. As a portion of the review effort described in this DSRS section and in DSRS Section 6.3, the appropriate reactor systems reviewer evaluates whether the entire break spectrum (break size and location) has been addressed; whether the appropriate break locations, break sizes, and initial conditions were selected in a manner that conservatively predicts the consequences of the LOCA for evaluating ECCS performance; and whether an adequate analysis of possible failure modes of ECCS equipment and the effects of the failure modes on the ECCS performance have been provided. For postulated break sizes and locations, the staff reviews the postulated initial reactor core and reactor system conditions, the postulated sequence of events including time delays prior to and after emergency power actuation, the calculation of the power, pressure, flow and temperature transients, the functional and operational characteristics of the reactor protection and ECCS systems in terms of how they affect the sequence of events, and operator actions required to mitigate the consequences of the accident.

A spectrum of LOCA break sizes is to be evaluated and the limiting break identified through sufficient analyses to determine the worst break peak clad temperature (PCT), the worst local clad oxidation, and the highest core wide oxidation percentage. The small break spectrum should have sufficient resolution to locate these limiting conditions. In the analysis of small pipe breaks, the effect of break size needs to be evaluated. The analyses must also be carried out until the top of the active fuel has been recovered with a two-phase mixture and the cladding temperatures have been reduced to temperatures near the saturation temperature. Break locations should include various locations around the letdown and charging piping to and from the CVCS. If operator action is required to maintain conditions within 10 CFR 50.46 limits, then the equipment and operator action times to achieve a successful core cooling condition should also be identified and evaluated.

An evaluation of post-LOCA long term cooling should also be performed to identify the operator actions to successfully control and prevent boric acid precipitation. Analyses of small break LOCAs should be performed to identify the timing for boric acid precipitation. A spectrum of small breaks should also be analyzed to identify other means to control boric acid precipitation when RCS pressure remains too high to enable flushing of the core. All equipment and operator action times should also be clearly identified in the analyses.

The calculational framework used for the evaluation of the ECCS system in terms of core short-term behavior and long-term cooling performance is referred to as an evaluation model. It includes one or more computer programs, the mathematical models used, the assumptions and correlations included in the program, the procedure for selecting and treating the program input and output information, the specification of those portions of the analysis not included in computer programs, the values of parameters, and all other information necessary to specify the calculational procedure. The evaluation model used by the applicant must comply with the acceptance criteria for ECCS given in 10 CFR 50.46. Should the LOCA blowdown calculations be modified for the purpose of studying structural behavior (for example, core support structure design, control rod guide structure design, steam generator design, containment design, reactor coolant system letdown and charging (CVCS) piping, and support structure design), all differences should be identified and described by the applicant. The reviewer evaluates these modifications, including analytical techniques, computer programs, values of input parameters, break size, type, and location, and all other pertinent information, and makes recommendations regarding their acceptability. The reviewer initiates a generic computer program review as required.

The staff review of this DSRS section covers the following areas:

A. The failure mode analysis of the ECCS to verify that an adequate analysis of possible failure modes of ECCS equipment and the effect of the failure modes on the ECCS performance has been provided in conjunction with the effort described in DSRS Section 6.3.

B. The analytical techniques and computer programs used by the applicant to determine the blowdown to the containment vessel, depressurization of the reactor vessel, condensation of steam on the containment vessel wall, and the opening of the RRV and RVV to maintain core water level above the active fuel.

C. The analytical techniques and computer programs used by the applicant for power transient calculations (including moderator temperature, void and fuel temperature, reactivity feedback effects, and decay heat) and for the cladding temperature, cladding rupture and swelling calculations.

D. Independent confirmatory calculations of the blowdown, reactor vessel depressurization, steam condensation, RRV and RVV opening to maintain core water level above the active fuel, and core water level refill phases of the accident, and cladding heatup calculations, as required to verify the applicant's conclusions.

E. Verification that the core physics data used by the applicant, or by the staff in independent audit analyses, is the appropriate data to be used.

F. The results of the post-LOCA long-term cooling analyses to assure that an acceptable model has been employed to identify the timing of boric acid precipitation for all break locations and sizes. The review will also verify that an adequate procedure has been devised to control boric acid precipitation for all breaks to assure long term cooling.

G. Containment peak pressure and heat transfer capacity to remove the decay heat are described in DSRS Section 6.2.2.

Steam generator tube rupture events shall also be reviewed as part of the LOCA break spectrum analysis. The reviewer shall review the potential coolant inventory loss from reactor vessel to the secondary side.

The reviewer provides an evaluation of fission product releases and radiological consequences. This effort is described in DSRS Section 15.0.3, "Design-Basis Accident Radiological Consequence Analyses for NuScale SMR Design."

2. <u>COL Action Items and Certification Requirements and Restrictions</u>. For a design certification (DC) application, the review will also address combined license (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.

<u>Review Interfaces</u>

Other DSRS and Standard Review Plan (SRP) sections interface with this section as follows:

1. General information on transient and accident analyses is provided in DSRS Section 15.0.

2. Design-basis radiological consequence analyses associated with design-basis accidents are reviewed under DSRS Section 15.0.3.

3. Fuel failure modes and burst correlations are evaluated for compliance with 10 CFR 50.46 as part of the fuel design review under SRP Section 4.2.

Evaluation of the functional capability of the containment for the spectrum of loss-of-coolant events is performed under DSRS Section 6.2.1. On request from the lead review of DSRS Section 15.6.5, the reviewer of the interfacing section verifies that the assumptions used for the containment response have been selected in a conservative manner for the period in which the maximum LOCA peak clad temperature occurs.

4. Aspects of the accident sequences described in the applicant's technical submittal are evaluated to determine whether the reactor and plant protection and safeguards controls and instrumentation systems will function as assumed in the safety analysis with regard to automatic actuation, remote sensing, indication, control, and interlocks with auxiliary or shared systems under DSRS Section 7.2. The reviewer evaluates the failure modes analysis of the ECCS to verify that the applicant has provided an adequate analysis of possible failure modes of ECCS instrumentation and controls equipment and the effect of the failure modes of that equipment on the ECCS performance.

5. Evaluation of the emergency onsite power functional capabilities is described in DSRS Sections 8.3.1 and 8.3.2. The staff verifies that the control systems power sources needed to function to mitigate the event are available as required by the applicant's description of the event. The staff reviewer evaluates the failure modes analysis of the ECCS to verify that the applicant has provided an adequate analysis of possible failure modes of ECCS equipment and the effect of the failure modes on the ECCS performance.

6. Evaluation of auxiliary systems (e.g., air system, hydraulic system for the operation of RVVs and RRVs, reactor pool water and refill capability) to confirm that these systems can supply all the functions required to support the ECCS in performing its function during and following a loss-of-coolant accident is reviewed in the applicable DSRS or SRP Sections in Chapter 9 and 10.

7. The effects of the combined blowdown and seismic loads on core support structures and on control rod guide structures are reviewed under SRP Sections 3.6.2, 3.9.2, 3.9.3, 3.9.4, and 3.9.5. The reviewer verifies that the core remains in a coolable geometry after a LOCA and that the control rods can also be inserted for breaks crediting this function. Analyses of the deformed bundle in the core should be performed to show that the acceptance criteria of 10 CFR 50.46 are met. The staff reviewer verifies that acceptable criteria have been employed in the design of the reactor coolant system and its supports to prevent failures of the reactor coolant pressure boundary and engineered safety feature equipment in the event of a LOCA.

8. Plant operating procedures are reviewed under SRP Section 13.5.2.1.

9. The determination of the risk significance of structures, systems, and components (SSCs) relied upon to meet required functions during the accidents are based on the review of the probabilistic risk assessment under SRP Chapter 19.

II. <u>ACCEPTANCE CRITERIA</u>

<u>Requirements</u>

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

1. 10 CFR 50.46 as it relates to ECCS equipment being provided that refills the vessel in a timely manner for a loss-of-coolant accident resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary.

2. 10 CFR Part 50, Appendix A, GDC 13, Instrumentation and control.

3. 10 CFR Part 50, Appendix A, GDC 35, Emergency core cooling.

4. 10 CFR Part 100, “Reactor Site Criteria,” as they relate to mitigating the radiological consequences of an accident.

5. 10 CFR Part 50, Appendix K, ECCS evaluation models.

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.

Specific criteria necessary to meet the relevant requirements of the regulations identified above and necessary to meet the Three Mile Island (TMI) Action Plan items are as follows:

1. An evaluation of ECCS performance has been performed by the applicant in accordance with an evaluation model that satisfies the requirements of 10 CFR 50.46. Regulatory Guide (RG) 1.157, “Best-Estimate Calculations of Emergency Core Cooling System Performance,” and Section I of Appendix K to 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities,” provide guidance and requirements on acceptable evaluation models. This also includes analyses of a spectrum of LOCAs to assure that boric acid precipitation is precluded for all break sizes and locations.

   The analyses must be performed in accordance with 10 CFR 50.46, including methods referred to in 10 CFR 50.46(a)(1) or (2). In accordance with GDC 35, the analyses must demonstrate sufficient redundancy in components and features, and must provide suitable interconnections, leak detection, isolation, and containment capabilities to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure. Additionally, the LOCA method used and the LOCA analyses should be shown to apply to the individual plant by satisfying 10 CFR 50.46(c)(2), and the analysis results must meet the performance criteria in 10 CFR 50.46(b):

   A. The calculated maximum fuel element cladding temperature does not exceed 1,200 degrees Celsius (C) (2,200 degrees Fahrenheit (F)).

   B. The calculated total local oxidation of the cladding does not exceed 17 percent of the total cladding thickness before oxidation. Total local oxidation includes pre-accident oxidation as well as oxidation that occurs during the course of the accident.

C. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam does not exceed 1 percent of the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

D. Calculated changes in core geometry are such that the core remains amenable to cooling.

E. After any calculated successful initial operation of the ECCS, the calculated core temperature is maintained at an acceptably low value and decay heat is removed for the extended period of time required by the long-lived radioactivity remaining in the core.

2. The radiological consequences of the most severe LOCA are within the guidelines of 10 CFR Part 100. Reviewers should use DSRS Section 15.0.3.

3. The TMI Action Plan items for II.E.2.3, II.K.3.30, and II.K.3.31 have been met.

4. Programmatic Requirements: 10 CFR 50.36(a)(2) requires that each design certification include proposed generic technical specifications (TS) for the portion of the plant that is within the scope of the design certification. The regulation 10 CFR 52.79(a)(30) requires that the COL applicant provide proposed technical specifications as part of its application. The applicant's analysis addressing the guidance in DSRS Section 15.6.5 must be consistent with the information presented in the applicant's TS.

<u>Technical Rationale</u>

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 10 CFR 50.46 requires that light-water-cooled nuclear power reactors be equipped with an emergency core cooling system designed so that core performance after postulated LOCAs conforms to specified criteria related to limiting core damage.

   The requirements specified in 10 CFR 50.46 provide an acceptable and conservative means of calculation of the consequences of LOCAs from a spectrum of pipe break sizes and locations that have been subject to careful review and experimental verification. If the calculations of the performance of the emergency core cooling system are conducted in accordance with these methods, there is a high level of probability that the acceptance criteria on core performance will not be exceeded and damage to the core and offsite consequences will be minimized. RG 1.157 and Appendix K to 10 CFR Part 50, provide guidance and requirements on evaluation models needed to demonstrate compliance with the acceptance criteria. Appendix K also specifies documentation required for evaluation models.

   Meeting the requirements and guidance outlined in the references provides assurance that following a LOCA the reactor core will remain in a coolable geometry and offsite consequences will be within the guidelines specified in 10 CFR Part 100.

2. GDC 13 requires the provision of instrumentation that is capable of monitoring variables and systems over their anticipated ranges for normal operation, for anticipated

operational occurrences, and for accident conditions, as appropriate, to assure adequate safety, and requires controls that can maintain these variables and systems within prescribed operating ranges.

GDC 13 applies to this section because the reviewer evaluates the sequence of events, including automatic actuations of protection systems, and manual actions, and determines whether the sequence of events is justified, based upon the expected values of the relevant monitored parameters and instrument indications.

3. Compliance with GDC 35 requires that a means of providing abundant emergency core cooling be provided that will transfer heat from the reactor core in the event of a LOCA. GDC 35 also requires the applicant to provide suitable redundancy of components and features, and suitable interconnections, leak detection, isolation, and containment capabilities, to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished assuming a single failure. GDC 35 specifies that an emergency core cooling system be installed in all nuclear power reactors. DSRS Section 15.6.5 specifies the analytical procedures that are to be followed to establish that the ECCS will function to meet acceptance criteria specified in 10 CFR 50.46.

   10 CFR Part 50, Appendix K and RG 1.157 provide requirements and guidance on calculational procedures needed to demonstrate compliance with the acceptance criteria.

   Meeting the requirements of GDC 35 will provide assurance that after a LOCA the reactor core will remain in a coolable geometry and offsite consequences will be within the guidelines specified in 10 CFR Part 100

4. The regulations in 10 CFR Part 100 describe criteria that guide the Commission in its evaluation of the suitability of proposed sites for nuclear power and testing reactors. Part 100 specifies radiation dose guidelines that should not be exceeded in the event of postulated accidents including LOCAs.

   To satisfy the requirements of 10 CFR Part 100 the applicant must demonstrate that the offsite doses resulting from various accidents presented in the applicant’s technical submittal are within the guideline values. Meeting the guideline doses is achieved by a combination of engineered safety features installed in the nuclear facility, an effective emergency core cooling system, and siting the nuclear plant in an area that does not exceed population density requirements.

   Meeting the nuclear power plant siting criteria provides a level of assurance that the plant will pose no undue risk to the public because of the consequences of LOCAs.

III. <u>REVIEW PROCEDURES</u>

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 procedures below are used during the standard DC and COL reviews, as appropriate. During the standard design certification review, the values of system parameter setpoints used in the analysis are considered preliminary in nature and subject to change. At the COL review, final values, if available, should be used in the analysis and the reviewer compares these to the limiting safety system settings included in the proposed technical specifications.

For the review of the ECCS performance analysis, as presented in the application the reviewer verifies 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. The calculations were performed using an evaluation model as specified in 10 CFR 50.46. The application should clearly state this and properly reference the evaluation model. If the analysis is done with a new evaluation model, a generic review of the new model is required. Evaluation models pertain to both the short term behavior following design bases breaks as well as post-LOCA long term cooling evaluations that properly address boric acid precipitation and prevention for LOCAs.

4. An adequate failure mode analysis has been performed to justify the selection of the most limiting single active failure. This analysis is reviewed in part under DSRS Section 6.3. If the design has been changed from that presented in previous applications, changes in the reactor coolant system, reactor core, and ECCS are reviewed with respect to the most limiting single failure.

5. A variety of break locations and the complete spectrum of break sizes were analyzed. If part of the evaluation is done by referencing earlier work, design differences (ECCS, reactor coolant system, reactor core, etc.) between the facilities in question are reviewed. If there are significant differences, the applicant should have made sensitivity studies on the important parameters. If such sensitivity studies are not presented in the applicant's technical submittal, the reviewer requests them.

6. If core uncovery is not expected during the entire period of a LOCA, the staff should ensure that a significant number of fuel rods will not be damaged because of local dryout conditions. This may be demonstrated by showing that the limiting fuel rod heat flux remains below the critical heat flux (CHF) at a given pressure after depressurization has taken place. If, however, the heat flux exceeds the CHF, further analyses should be performed to estimate the amount of fuel damage expected from "burn-out" while the bulk of the core remains covered with water during the LOCA. Fuel damage and potential for radioactivity release to the environment must be consistent with 10 CFR Part 100. If such evaluations are not provided in the applicant's technical submittal, the reviewer requests that they be made.

7. The parameters and assumptions used for the calculations were conservatively chosen, including the following points:

A. The initial power level is taken as the licensed core thermal power initially assumed to be operating plus an allowance of 2 percent to account for power measurement uncertainties, unless a lower level of uncertainty can be justified by the applicant. The operating condition at the initiation of the event should correspond to the operating condition that maximizes the consequences of the event.

B. The maximum linear heat generation rate used should be based on the proposed licensed core thermal power as discussed in Item A and the technical specification limit on peaking factors, or the technical specification limits on maximum linear heat generation rate. For many plants these limits may also be found in Technical Specification 5.6.5, Core Operating Limits Report (COLR).

C. All permitted axial power shapes, as given in Section 4.3 of the SRP, should be addressed by the analyses. Normally, the evaluation model will identify the least favorable axial shape as a function of break size. If the evaluation model did not discuss axial shapes, or the discussion is not applicable to a given case, sensitivity studies are requested.

D. The initial stored energy was conservatively calculated by the applicant. The value used is checked against the applicant's steady-state temperatures, as given in DSRS Section 4.4, similar calculations performed by the staff, or calculations done for similar plants by previous applicants.

E. Appropriate analyses are presented to support any credit taken for control rod insertion.

F. The analysis of boric acid precipitation should include a justified mixing volume, which is computed as a function of time as ECC injection enters the core region. The precipitation limit must also be justified in the evaluation model. If the system design includes high concentrate boric acid tanks and/or sources, then these systems must be assumed to be operating at the time of the break initiation.

G. The containment pressure response used during the ECCS performance evaluation reflects a justified conservative initial containment pressure. Debris wash-down into the bottom of the containment has been adequately treated so as to result in a conservative recirculation flow rate after sufficient water accumulates in containment.

8. Reactor protection system actions, and the actuation of the RRVs and RVVs, are consistent with the set points and the associated uncertainties and delay times listed in the applicant's technical submittal. The ECCS flow rates should be checked against the applicant's data on natural circulation flow that are sufficiently similar to the NuScale ECCS system. The Regional Offices may be requested to provide data of this type from the startup tests for new designs and from periodic tests on duplicate designs.

   In the case of NuScale, which uses passive rather than active systems to provide ECCS to the reactor vessel, pressure drop test results should be reviewed to determine that the passive ECCS system characteristic is consistent with that in the analyses of the system performance.

9. The results of the applicant's calculations are reasonable based on the selected input parameters. The following variables should be reviewed on a generic basis and spot-checked thereafter: power transients for various breaks; pressure transients at various system locations; flow transients near the break, in the core, and in the downcomer; reactor coolant temperature and quality at core inlet, core outlet, and in-core; cladding temperature transients (core average, hot assembly, hot pin); heat transfer coefficients during blowdown, depressurization, refill; heat flux transients from piping and vessel walls; primary-secondary heat transfer; timing of clad rupture (if the peak clad temperature could be appreciably higher when perforation occurs at a different but equally probable time, calculations with modified assumptions are requested); and peak clad temperature as a function of break size (if it is uncertain whether the peak value has been found, additional calculations are requested). The boric acid concentration should be shown as a function of time for the limiting small breaks.

   NuScale may base its ECCS and reactor coolant system designs on prevention of core uncovery. Should that be the case, the reviewer should compare the applicant's analysis with the staff's independent analysis to determine if the predicted level of core coverage is consistent.

10. The calculated peak clad temperature, maximum local oxide thickness, and core average zirconium-water reaction meet the acceptance criteria for ECCS given in 10 CFR 50.46(b). Boric acid concentration should be shown to be controlled before reaching the precipitation limit and all equipment and operator action times identified to prevent boron precipitation should be included in the EOPs.

11. The applicant's analysis addresses the full LOCA sequence of events, for the full spectrum of break sizes and locations, to the point where the plant is in the long-term cooling mode and removal of decay heat has been well established for small breaks. The reviewer checks the assumed sources of coolant water, redundancy of delivery routes, alignment of valves, control of boron concentration and all required operator actions.

12. The TMI Action Plan items are reviewed to assure compliance with the acceptance criteria.

    A. The reviewer evaluates the uncertainty analyses performed by the applicant to assure that the modeling assumptions and phenomena for small-break LOCA calculations are properly accounted for to determine the acceptability of the ECCS performance pursuant to 10 CFR 50.46 and RG 1.157 or Appendix K of 10 CFR Part 50 (Item II.E.2.3)

    B. The reviewer evaluates the small-break LOCA model verification performed by the applicant and assures that any modifications required are incorporated into the specific plant analyses (Item II.K.3.30 and II.K.3.31).

13. Reviewers representing all relevant technical disciplines will provide input for the areas of review stated in Subsection I. The staff review uses such input as required to assure that this review procedure is complete.

14. Emergency preparedness and radiation protection reviewers assess fission product releases and radiological consequences of design-basis (most severe) LOCAs as part of the DSRS Section 15.0.3 review, based on the application.

15. 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 applicant's 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., site suitability report or topical report).

IV. EVALUATION FINDINGS

The reviewer verifies that the applicant has provided sufficient information and that the staff's technical review and analysis, as augmented by the application of programmatic requirements in accordance with the staff's technical review approach in the NUREG-0800 Introduction Part 2, 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 that the loss-of-coolant analysis resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary is acceptable and meets the relevant requirements of 10 CFR 50.46, GDC 13, GDC 35, and 10 CFR Part 100. This conclusion is based on the following:

1. The applicant meets GDC 13 requirements by demonstrating that all credited instrumentation will be available to monitor variables and systems over their anticipated ranges, and that actuations of protection systems, automatic and manual, occur at values of monitored parameters to ensure that the variables and systems are within their prescribed operating ranges.

2. The applicant meets GDC 35 requirements by demonstrating that a means of providing abundant emergency core cooling is provided that will transfer heat from the reactor core in the event of a LOCA. The applicant also demonstrated that it meets the GDC 35 requirement to provide suitable redundancy of components and features, and suitable interconnections, leak detection, isolation, and containment capabilities, to assure for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished assuming a single failure. Meeting the requirements of GDC 35 will provide assurance that after a LOCA the reactor core will remain in a coolable geometry and offsite consequences will be within the guidelines specified in 10 CFR Part 100.

3. The applicant has performed analyses of the performance of the ECCS, considered a spectrum of postulated break sizes and locations, and used an evaluation model that meets the requirements of 10 CFR 50.46. The results of the analyses show that the ECCS satisfies the following criteria:

    A. The calculated maximum fuel rod cladding temperature does not exceed 1,200 degrees C (2,200 degrees F).

B. The calculated total maximum local oxidation of the cladding does not exceed 17 percent of the total cladding thickness before oxidation.

C. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam does not exceed 1 percent of the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

D. Calculated changes in core geometry are such that the core remains amenable to cooling.

E. After any calculated successful initial operation of the ECCS, the calculated core temperature is maintained at an acceptably low value and decay heat is removed for the extended period of time required by the long-lived radioactivity remaining in the core.

F. The applicant has satisfied the relevant TMI Action Plan items.

G. Boric acid precipitation can be prevented for all break sizes and locations during post-LOCA long term cooling.

4. The radiological consequences meet 10 CFR 100 requirements for the postulated spectrum of LOCAs, which were evaluated from the viewpoint of site acceptability. For the purposes of this analysis, large fractions of the fission products were assumed to be released from the core even though these releases would be precluded by the performance of the ECCS.

The staff concludes that the calculated performance of the ECCS after a postulated LOCA and the conservatively calculated radiological consequences of such an accident conform to the Commission's regulations and to applicable regulatory guides and staff technical positions and, accordingly, the ECCS is considered acceptable.

For DC and COL reviews, the findings will also summarize the staff's evaluation regarding DC requirements and restrictions (e.g., interface requirements and site parameters) and COL action items relevant to this DSRS section.

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,* "Standards for Protection against Radiation," Part 20, Chapter I, Title 10, "Energy."

2. *U.S. Code of Federal Regulations,* "Domestic Licensing of Production and Utilization Facilities," Part 50, Chapter I, Title 10, "Energy."

3. *U.S. Code of Federal Regulations,* "Licenses, Certifications, and Approvals for Nuclear Power Plants," Part 52, Chapter I, Title 10, "Energy."

4. U.S. Nuclear Regulatory Commission, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident," Regulatory Guide (RG) 1.7, Revision 3, March 2007, Agencywide Documents Access and Management System (ADAMS) Accession No. ML070290080.

5. U.S. Nuclear Regulatory Commission, "Calculations of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors," RG 1.112, Revision 1, March 2007, ADAMS Accession No. ML070320241.

6. U.S. Nuclear Regulatory Commission, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," RG 1.183, July 2000, ADAMS Accession No. ML003716792.

7. American National Standards Institute/American Nuclear Society Standard 18.1-1999, "Source Term Specification."

8. U.S. Nuclear Regulatory Commission, “Clarification of TMI Action Plan Requirements,” NUREG-0737, ADAMS Accession No. ML051400209.

9. *U.S. Code of Federal Regulations,* “Environmental Radiation Protection Standards for Nuclear Power Operations,” Part 190, Title 40, “Protection of the Environment.”

10. U.S. Nuclear Regulatory Commission, “Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants,” RG 1.89, Revision 1, June 1984, ADAMS Accession No. ML003740271.

11. U.S. Nuclear Regulatory Commission, “Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants” RG 1.143, Revision 2, November 2001, ADAMS Accession No. ML013100305.

12. U.S. Nuclear Regulatory Commission, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants,” RG 1.26, Revision 4, March 2007, ADAMS Accession No. ML070290283.

13. U.S. Nuclear Regulatory Commission, “Seismic Design Classification,” RG 1.29, Revision 4, March 2007, ADAMS Accession No. ML070310052.

14. U.S. Nuclear Regulatory Commission, “Tornado Design Classification,” RG 1.117, Revision 1, April 1978, ADAMS Accession No. ML003739346.

15. U.S. Nuclear Regulatory Commission, “Combined License Applications for Nuclear Power Plants (LWR Edition),” RG 1.206.

16. Electric Power Research Institute, “Pressurized Water Reactor Primary Water Chemistry Guidelines.”

17. Electric Power Research Institute, “Pressurized Water Reactor Primary Water Zinc Application Guidelines.”

18. Electric Power Research Institute, “Advanced Light Water Reactor Utility Requirements Document, Volume III, ALWR Passive Plant.”

19. U.S. Nuclear Regulatory Commission, “NRC Review of Electric Power Research Institute’s Advanced Light Water Reactor Utility Requirements Document, Passive Plant Designs” NUREG-1242, Volume 3, Part 1 and Volume 3, Part 2 (ADAMS Accession Nos. ML070600372 and ML070600373).

20. Electric Power Research Institute, “Cobalt Reduction Guidelines.”

21. U.S. Nuclear Regulatory Commission, “Information Relevant to Assuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as is Reasonably Achievable,” RG 8.8, Revision 3, June 1978, ADAMS Accession No. ML003739549.