Draft for Final

U.S. NUCLEAR REGULATORY COMMISSION

DESIGN-SPECIFIC REVIEW STANDARD FOR NuScale SMR REACTOR DESIGN

6.2.1.4 MASS AND ENERGY RELEASE ANALYSIS FOR POSTULATED SECONDARY SYSTEM PIPE RUPTURES

REVIEW RESPONSIBILITIES

Primary - Organization responsible for the review of containment integrity

Secondary - None

I. AREAS OF REVIEW

The NuScale reactor containment is a low alloy steel evacuated vessel that surrounds the reactor vessel and is immersed in a large borated-water reactor building pool that serves as the Ultimate Heat Sink (UHS) for the passive Emergency Core Cooling System (ECCS), the passive Containment Heart Removal System (CHRS), and the passive Decay Heat Removal System (DHRS) heat exchangers located in the reactor building pool attached to the outside 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.

Each containment vessel is penetrated on opposite sides of the vessel by a feedwater line and steam line so that two separate helical coil steam generators (HCSGs) are serviced separately in the reactor vessel. The isolation valves for the feedwater and main steam lines and the safety and relief valves for the main steam lines are all located outside and above the containment vessel for each reactor module. The redundant DHRS heat exchangers' piping is downstream of the feedwater isolation valve and upstream of the main steam isolation valve. The DHRS is actuated by closure of the feedwater and main steam isolation valves and the opening of one of two parallel heat exchanger condensate valves to permit natural circulation flow of the always-flooded feedwater inventory of the DHRS heat exchanger to the HCSG to provide heat removal from the reactor to the reactor building pool. Because of the close proximity of the enclosing containment vessel to the reactor vessel, a break in either of the two feedwater lines or the two main steam lines would occur in a short run of piping between the nozzle at the reactor vessel and the penetration fixture on the containment vessel.

An in-containment break of either a feedwater line or a main steam line will be limited upon closure of the feedwater and main steam isolation valves to the inventory of feedwater and steam in the short run of piping and the inventory in one of the two HCSG’s plus, if the DHRS condensate valves open, the content of cold water in the flooded DHRS heat exchanger on the break-affected line. The evacuated containment vessel will accommodate the sudden pressure increase, and the passive heat dump provided by the surrounding reactor building pool will both cool and condense steam entering the evacuated containment and cool the shutdown reactor through natural circulation flow in the reactor and on the secondary side through the unaffected

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HCSG and DHRS heat exchanger assuming that the isolation valves close on the unaffected secondary loop and that shutdown reactor heat is not still being dumped to the main condenser on that loop. If the break is in the in-containment portion of the main steam line, steam will be dumped into containment until the inventory in the affected DHRS heat exchanger is emptied by gravity drain of the elevated DHRS heat exchanger inventory into the HCSG on the affected loop. If the break is on the main feedwater piping in-containment, flashing steam from heated water in the feedwater line downstream of the isolation valve and the HCSG will first empty into the containment followed by cold water from the elevated DHRS heat exchanger on the affected loop In either scenario, the irradiated containment vessel being pressurized by hot steam from an initial zero evacuated pressure condition while sitting in cold pool water will have to be evaluated. The overall effect of the pressure spike on containment vessel integrity will have to demonstrate adequate margin in containment integrity (typically by at least 10 percent).

The mass and energy release analysis for secondary system pipe ruptures is reviewed to ensure the acceptability of the data used to evaluate the containment vessel functional design.

The specific areas of review are as follows:

1. Sources of Energy: All of the energy sources from steam and feedwater line break accidents that are available for release to the containment are reviewed.

2. Mass and Energy Release Rate: The mass and energy release rate calculations are reviewed.

3. Single-Failure Analyses: The single-failure analyses performed for steam and feedwater line isolation provisions that would limit the flow of steam or feedwater to the assumed pipe rupture are reviewed.

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 various aspects of the containment design are identified in DSRS Section 6.2.1.1.A.

2. The seismic classification and system quality group classification of steam and feedwater line isolation valves are reviewed under SRP Sections 3.2.1 and 3.2.2 to determine the acceptability of these valves in limiting the mass and energy releases from the steam and feedwater systems.

3. Postulated pipe break locations and sizes are reviewed under SRP Section 3.6.2.

4. Risk significance of SCCs is reviewed under SRP Section 19.0 and 19.3.

II. ACCEPTANCE CRITERIA

Requirements

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

1. General Design Criteria (GDC) 50, as it relates to providing sufficient conservatism in the mass and energy release analysis for postulated secondary system pipe ruptures to ensure the reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident.

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 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.

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.

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. Sources of Energy. The sources of energy that should be considered in the analyses of steam and feedwater line break accidents include the stored energy in the affected HCSG's metal, including the vessel tubing, feedwater line, and steam line; stored energy in the water contained within the affected HCSG; stored energy in the feedwater transferred to the affected HCSG before closure of the isolation valves in the feedwater line; stored energy in the steam from the unaffected HCSG before the closure of the isolation valves in the HCSG crossover lines; and energy transferred from the primary coolant to the water in the affected HCSG during blowdown to include energy transferred to the draining DHRS heat exchanger water.

   The steam line break accident should be analyzed for a spectrum of pipe break sizes and various plant conditions from hot standby to 102 percent of full power. The applicant need only analyze the 102-percent power condition if it can demonstrate that the feedwater flows and fluid inventory are greatest at full power.

2. Mass and Energy Release Rate. In general, calculations of the mass and energy release rates during a steam or feedwater line break accident should be performed in a conservative manner from a containment response standpoint (i.e., the post-accident containment pressure and temperature are maximized). The following criteria indicate the degree of conservatism that is desired:

   A. Mass release rates should be calculated using the Moody model (Ref. 6) for saturated conditions or a model that is demonstrated to be equally conservative.

   B. Calculations of heat transfer to the water in the affected HCSG should be based on nucleate boiling heat transfer.

   C. Calculations of mass release should consider the water in the affected HCSG and feedwater line, feedwater transferred to the affected HCSG before the closure of the isolation valves in the feedwater lines and upon flooding with the DHRS heat exchanger inventory in the affected loop, and steam in the HCSG,

   D. If liquid entrainment is assumed in the steam line breaks, experimental data should support the predictions of the liquid entrainment model. A spectrum of steam line breaks should be analyzed, beginning with the double-ended break and decreasing in area until no entrainment is calculated to occur. This will allow selection of the maximum release case.

      If no liquid entrainment is assumed, a spectrum of the steam line breaks should be analyzed beginning with the double-ended break and decreasing in area until it has been demonstrated that the maximum release rate has been considered.

   E. Feedwater flow to the affected HCSG should be calculated considering the diversion of flow from the other HCSG between the two feedwater pipes to the common header with inlets to the HCSG on opposite sides of the reactor vessel, feedwater flashing, and increased feedwater pump flow caused by the reduction

in HCSG pressure. An acceptable method for computing feedwater flow is to assume all feedwater travels to the HCSG at the pump run-out rate before isolation. After isolation, the unisolated feedwater mass should be added to the available inventory in the HCSG.

Any general-purpose thermal-hydraulics computer codes that the responsible reviewing organization for the subject application finds acceptable may be used to compute mass and energy releases from steam and feedwater line break accidents.

3. Single-Failure Analyses. Steam and feedwater line break analyses should assume a single active failure in the steam or feedwater line isolation provisions to maximize the containment peak pressure and temperature.

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 systems be designed to accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss of coolant. DSRS Section 6.2.1.4 applies the requirements of this GDC to postulated secondary system pipe ruptures to assure that mass and energy inputs are appropriately conservative. A secondary system pipe rupture releases a significant amount of energy which potentially could damage the containment structure or associated systems. Containment, therefore, must be designed to definitively withstand this accident. Meeting the requirements of GDC 50 will ensure that containment integrity is maintained under the most severe secondary system pipe rupture, 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: Integral Pressurized Water Reactor Edition" (NUREG-0800 Intro Part 2) as applied to this DSRS Section, the staff will review the information proposed by the applicant to evaluate whether it meets the acceptance criteria described in Subsection II of this DSRS. As noted in NUREG-0800 Intro Part 2, the NRC requirements that must be met by an SSC do not change under the SMR framework. Using the graded approach described in NUREG-0800 Intro Part 2, the NRC staff may determine that, for certain structures, systems, and components (SSCs), the applicant's basis for compliance with other selected NRC requirements may help demonstrate satisfaction of the applicable acceptance criteria for that SSC in lieu of detailed independent analyses. The design-basis capabilities of specific SSCs would be verified

where applicable as part of completion of the applicable ITAAC. The use of the selected programs to augment or replace traditional review procedures is described in Figure 1 of NUREG-0800, Introduction - Part 2. Examples of such programs that may be relevant to the graded approach for these SSCs include:

- 10 CFR Part 50, Appendix A, General Design Criteria (GDC), Overall Requirements, Criteria 1 through 5
- 10 CFR Part 50, Appendix B, Quality Assurance (QA) Program
- 10 CFR 50.49, Environmental Qualification of Electrical Equipment (EQ) Program
- 10 CFR 50.55a, Code Design, Inservice Inspection and Inservice Testing (ISI/IST) Programs
- 10 CFR 50.65, Maintenance Rule requirements
- Reliability Assurance Program (RAP)
- 10 CFR 50.36, Technical Specifications
- Availability Controls for SSCs Subject to Regulatory Treatment of Non-Safety Systems (RTNSS)
- Initial Test Program (ITP)
- Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC)

This list of examples is not intended to be all-inclusive. It is the responsibility of the technical reviewers to determine whether the information in the application, including the degree to which the applicant seeks to rely on such selected programs and guidance, demonstrates that all acceptance criteria have been met to support the safety finding for a particular SSC.

2. In accordance with 10 CFR 52.47(a)(8),(21), and (22), and 10 CFR 52.79(a)(17), (20) and (37), for design certification or combined license applications submitted under Part 52, the applicant is required to (1) address the proposed technical resolution of unresolved safety issues and medium- and high-priority generic safety issues which are identified in the version of NUREG-0933 current on the date up to 6 months before the docket date of the application and which are technically relevant to the design; (2) demonstrate how the operating experience insights have been incorporated into the plant design; and, (3) provide information necessary to demonstrate compliance with any technically relevant portions of the Three Mile Island requirements set forth in 10 CFR 50.34(f), except paragraphs (f)(1)(xii), (f)(2)(ix), and (f)(3)(v) for a DC application, and except paragraphs (f)(1)(xii), (f)(2)(ix), (f)(2)(xxv), and (f)(3)(v) for a COL application. These cross-cutting review areas should be addressed by the reviewer for each technical subsection and relevant conclusions documented in the corresponding safety evaluation report (SER) section.

3. <u>Sources of Energy</u>. The reviewer evaluates the sources of energy identified by the applicant in the analyses of steam and feedwater line break accidents to ensure that the sources listed in Subsection II of this DSRS section have been considered.

   The reviewer also examines the assumptions of the secondary coolant system pipe break analysis to determine whether the applicant has identified the worst case pipe break accident and completed the analysis in a conservative manner from the standpoint

of containment pressure and temperature. This review involves the proposed methods and models used for blowdown analyses. The reviewer will evaluate the acceptability of the approach used by the applicant based on the acceptance criteria in Subsection II of this DSRS section.

4. Mass and Energy Release Rate. The reviewer evaluates the applicant's calculations for main feedwater flow into the HCSG to determine whether the flow rate is conservatively maximized.

   If the applicant's steam line break model calculates liquid entrainment, the reviewer determines the validity of the experimental data provided to support the entrainment calculation. The reviewer evaluates comparisons to experimental data made by the applicant and makes comparisons to other available experimental data to determine the amount of conservatism in the mass and energy release models.

   The reviewer examines the results of a spectrum of steam line breaks, beginning with the double-ended break and decreasing in area until no entrainment occurs, to ensure that the applicant has identified the steam line break size producing the highest containment temperature and pressure.

   The reviewer may perform confirmatory analyses of the containment pressure and temperature response to steam and feedwater line breaks inside the containment using thermal-hydraulic computer codes that the responsible reviewing organization for the subject application finds acceptable.

5. Single-Failure Analyses. The reviewer reviews analyses of postulated single failures of active components in the secondary systems, such as steam and feedwater line isolation valves and feedwater pumps, and determines whether the single failure that maximizes containment pressure and temperature has been selected.

   The reviewer requests the review of SRP Sections 3.2.1, 3.2.2, and 3.6.2 by the responsible organization as to the acceptability of non-safety valves in limiting the mass and energy releases from the steam and feedwater systems.

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

The evaluation findings will follow the format provided in DSRS Section 6.2.1.1.A and conclude that the applicant followed the DSRS acceptance criteria identified above or identified deviations from the DSRS acceptance criteria with appropriate justification and meets GDC 50 as it relates to providing sufficient conservatism in the mass and energy release analysis for postulated secondary system pipe ruptures for the containment design basis.

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 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 Guide 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)."

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

6. F. J. Moody, "Maximum Flow Rate of a Single Component, Two-Phase Mixture," with discussion comments and Authors' Closure, Jour. of Heat Transfer, Trans. Am. Soc. of Mechanical Engineers, Vol. 87, No. 1, February 1965.