U.S. NUCLEAR REGULATORY COMMISSION

DESIGN-SPECIFIC REVIEW STANDARD for NuScale SMR DESIGN

15.1.5 STEAM SYSTEM PIPING FAILURES INSIDE AND OUTSIDE OF CONTAINMENT

REVIEW RESPONSIBILITIES

Primary - Organization responsible for the review of transient and accident analyses

Secondary - None

I. <u>AREAS OF REVIEW</u>

The steam release resulting from a rupture of a main steam pipe will cause an increase in steam flow, which decreases with time as the steam pressure decreases. The increased steam flow causes increased energy removal from the reactor coolant system (RCS) and results in a reduction of coolant temperature and pressure. The negative moderator temperature reactivity feedback and the cooldown of the reactor system cause an increase in core reactivity. The core reactivity increase could cause a loss of reactor core shutdown margin and a resulting increase in reactor power. If the plant is at power, the reactor is automatically tripped and the steam generator is isolated. The decay heat removal system (DHRS) automatically activates and decay heat is removed from the RCS to the ultimate heat sink.

Analysis of the transient after a steam line break is sensitive to the fluid discharge rate at the break so that a range of break sizes must be considered both inside and outside containment to determine the acceptability of the system response. The course that the transient takes and its ultimate effects also depend on the assumed initial power level and mode of operation (e.g., hot shutdown; full power). Evaluation with various assumed initial conditions is required to verify that the condition leading to the severest consequences has been identified.

The specific areas of review are as follows:

1. postulated initial core and reactor conditions pertinent to the steam line break accident

2. methods of thermal and hydraulic analyses, including the effects of potential hydraulic instabilities

3. postulated sequence of events, including analyses to determine the time of reactor trip and time delays prior to and subsequent to initiation of the reactor protection system

4. assumed responses of the reactor coolant and auxiliary systems

5. functional and operational characteristics of the reactor protection system in terms of its effects on the sequence of events

6. operator actions required to secure and maintain the reactor in a safe-shutdown condition

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7. core power excursion because of power demand created by excessive steam flow out of the break

8. variables influencing neutronics

The results of the analyses are reviewed to ensure that pertinent system parameters are within expected ranges. The parameters of importance for these transients include:

A. RCS pressure
B. steam generator pressure
C. fluid temperatures
D. clad temperatures
E. discharge flow rates
F. steam line and feedwater flow rates
G. safety and relief valve flow rates
H. pressurizer and steam generator water levels
I. reactor power
J. total core reactivity
K. hot and average channel heat flux
L. minimum departure from nucleate boiling ratio (DNBR)

The sequence of events described in the applicant's technical submittal is reviewed by the branches that review reactor systems and instrumentation and control. The reviewer from the branch that reviews reactor systems concentrates on the capability of the reactor protection system, the engineered safety systems, and operator action to secure and maintain the reactor in a safe condition.

The analytical methods are reviewed by the branch that reviews reactor systems to find out whether the mathematical modeling and computer codes have been reviewed and accepted by the staff. If a referenced analytical method has not been reviewed, Reactor Systems starts an evaluation of the new analytical model.

9. Combined License (COL) Action Items and Certification Requirements and Restrictions. For a design certification (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 design-specific review standard (DSRS) sections interface with this section as follows:

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

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

3. SRP Sections 4.2, 4.3 and DSRS Section 4.4. Values of the parameters in the analytical models of the reactor core are reviewed for compliance with plant design and specified operating conditions, acceptance criteria for fuel cladding damage limits are determined, and the core physics, fuel design, and core thermal-hydraulics data in the technical submittal analysis are reviewed under SRP Sections 4.2, 4.3 and DSRS Section 4.4.

4. Standard Review Plan (SRP) Section 3.9.3. The DHRS system is reviewed to verify its ability to function after a steam line break given a single active component failure with either onsite or offsite power under SRP Section 5.4.7.

5. SRP Sections 3.6.2, and 3.9.1 through 3.9.3.evalute the effects of blow-down loads, including jet propulsion piping and component supports and the design bases for safety and relief valves. Design bases for safety and relief valves are also reviewed under SRP Section 3.9.3.

6. SRP Section 5.2.3 and DSRS Section 5.3.1. Fracture toughness properties of the reactor coolant pressure boundary and reactor vessel are reviewed under SRP Section 5.2.3 and DSRS Section 5.3.1.

7. DSRS Section 6.2.1. The response of the containment to ruptures of steam lines with regard to the effects of pressure and temperature on the containment functional capabilities is reviewed under DSRS Section 6.2.1. Analytical methods for deriving mass energy releases exiting the postulated break are reviewed under DSRS Section 6.2.1.4.

8. DSRS Sections 7.0 through 7.2. Aspects of the sequence described in the technical submittal are reviewed to evaluate 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. This review includes the instruments and controls required to ensure automatic and manual ECCS, or DHRS initiation and flow indication in the control room and is performed under DSRS Sections 7.0 through 7.2. The potential bypass modes and the possibility of manual control by the operator are also reviewed under DSRS Sections 7.0 through 7.2.

II. ACCEPTANCE CRITERIA

Requirements

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

The general objective of the review of steam line rupture events is to verify that short-term and long-term ability to cool the core has been achieved by confirming that the primary RCS is maintained in a safe status for a break equivalent in area to the double-ended rupture of the largest steam line. The acceptance criteria are based on meeting the relevant requirements of the following regulations:

1. General Design Criterion (GDC) 13, “Instrumentation and Control”

2. GDC 17, “Electric Power Systems”

3. GDC 27, "Combined Reactivity Control Systems Capability"

4. GDC 28, "Reactivity Limits"

5. GDC 31, "Fracture Prevention of Reactor Coolant Pressure Boundary"

6. Requirements for ensuring adequate decay heat removal are specified in Title 10 of the *Code of Federal Regulations* (CFR), Section 50.34(f)(2)(xii) [1].

<u>DSRS Acceptance Criteria</u>

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 above regulations are as follows:

1. Pressure in the reactor coolant and main steam systems should be maintained below acceptable design limits, considering potential brittle, as well as ductile failures.

2. The potential for core damage is evaluated on the basis that it is acceptable if the minimum DNBR remains above the 95/95 DNBR limit for pressurized-water reactors (PWRs) based on acceptable correlations (see DSRS Section 4.4). If the DNBR falls below these values, fuel failure (rod perforation) must be assumed for all rods that do not meet these criteria unless it can be shown, based on an acceptable fuel damage model (see SRP Section 4.2), which includes the potential adverse effects of hydraulic instabilities, that fewer failures occur. Any fuel damage calculated to occur must be of sufficiently limited extent that the core will remain in place and intact with no loss of core cooling capability.

3. The radiological criteria used in the evaluation of steam system pipe break accidents (PWRs only) appear in DSRS Section 15.0.3.

4. System(s) provided for decay heat removal must be highly reliable and, when required, automatically initiated. For the NuScale design, the DHRS provides the safety-related means of decay heat removal.

There are certain assumptions regarding important parameters used to describe the initial plant conditions and postulated system failures which should be used. These are listed below:

1. The reactor power level, other plant operating mode parameters assumed at the initiation of the transient should correspond to the operating condition that maximizes the consequences of the accident. Sensitivity studies will be required to determine the most

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[1] For 10 CFR Part 50 applicants not listed in 10 CFR 50.34(f), the applicable provisions of 10 CFR 50.34(f) will be made a requirement during the licensing process.

conservative combination of power level and plant operating mode. These sensitivity studies may be presented in a generic report and referenced in the technical submittal.

2. Assumptions as to the loss of offsite power (LOOP) and the time of loss should be made to study their effects on the consequences of the accident. A LOOP could occur simultaneously with the pipe break or during the accident, or offsite power might not be lost. Analyses should be made to determine the most conservative assumption appropriate to the particular plant design. The reviewer should note that the assumption that offsite power is not lost may maximize heat removal from the core and reactor system and thereby maximize containment pressure and reactivity feedback within the core. The analyses should take account of the effect that LOOP has on the main feedwater pump trips, on the initiation of the decay heat removal system and the effects on the sequence of events for these accidents. For new applications, LOOP should be considered in addition to any limiting single active failure. (This position is based upon interpretation of GDC 17, as documented in the final safety analysis report (FSAR) for the ABB-CE System 80+ DC.)

3. The effects (pipe whip, jet impingement, reaction forces, temperature, humidity, etc.) of postulated steam line breaks on other systems should be considered in a manner consistent with the intent of Branch Technical Position (BTP) 3-3 and BTP 3-4.

4. The worst single active component failure should be assumed to occur. For new applications, LOOP should not be considered as a single failure, (see Assumption 2 above). The assumed single failure could cause the steam generator to blow down, failure of main feedwater to isolate, or could be in any of the systems required to control the transient.

5. The maximum-worth control rod should be assumed to be held in the fully withdrawn position. An appropriate rod reactivity worth versus rod position curve should be used. Local power peaking at the location of the stuck out control rod should be considered. Local power peaking will affect the DNBR analysis in the initial period as the safety rods are entering the core and during any subsequent return to power resulting from reactivity addition to the core from the cooldown.

6. The core burnup (time in core life) should be selected to yield the most limiting combination of moderator temperature reactivity feedback, void reactivity feedback, Doppler reactivity feedback, axial power profile, and radial power distribution.

7. The initial core flow assumed for the analysis of the steam line rupture accident should be chosen conservatively. If the minimum core flow allowed by the technical specifications is assumed, the minimum DNBR margin results; however, for the analysis of steam line break accidents, this may not be the most conservative assumption. For example, maximum initial core flow results in increased RCS cooldown and depressurization, decreased shutdown margin, and an increased possibility that the core will become critical and return to power. Because it is not clear what initial core flow is most conservative, the assumed value should be justified.

8. For postulated pipe failure in nonseismically qualified portions of the main steam line (outside containment and downstream of the main steam isolation valves (MSIVs) because of a seismically initiated event, only safety-related equipment should be assumed operative to mitigate the consequences of the break.

9. For postulated instantaneous pipe failures in seismically qualified portions of the main steam line (inside containment and upstream of the MSIVs), only safety-related equipment should be assumed operative. If, in addition, a single malfunction or failure of an active component is postulated, credit may be taken for the use of a backup nonsafety-related component to mitigate the consequences of the break.

10. During the initial 10 minutes of the transient, should credit for operator action be required, an assessment for the limiting consequence must be performed to account for operator delay or error.

<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 GDC 13 requires the provision of instrumentation to monitor variables and systems over their anticipated ranges and of appropriate controls that can maintain listed 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.

2. Compliance with GDC 17 requires in part that onsite and offsite electrical power systems be provided to permit functioning of structures, systems, and components (SSCs) important to safety. The safety function for each power system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to ensure that (1) the specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences, and (2) the core is cooled and containment integrity and other vital functions are maintained.

   GDC 17 is applicable to this section because it requires that the LOOP be considered not as a single failure event, but assumed in the analyses for each event without changing the event category. Thus, the applicant should consider a LOOP concurrent with a single failure in the analysis of steam system piping failures.

3. Compliance with GDC 27 requires reactivity control systems to be designed to have a combined capability (in conjunction with poison added by the emergency core cooling (ECC) system) of reliably controlling reactivity changes, thereby ensuring that the capability for core cooling is maintained under postulated accident conditions and with appropriate margin for stuck rods.

   Compliance with GDC 28 requires reactivity control systems to be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding, nor (2) sufficiently disturb the core, its support structures, or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor temperature and pressure, and addition of cold water.

GDCs 27 and 28 are applicable to this section because the reviewer evaluates steam system piping failures, both inside and outside containment that could cause transient conditions affecting reactor coolant temperature and pressure, including complex changes in core reactivity. The applicant's analyses of these transients in the technical submittal must demonstrate that reactivity, pressure, and temperature changes will not be severe enough to cause an unacceptable effect on the reactor coolant pressure boundary or on the capability for cooling the core. These analyses must be independently reviewed by the staff in accordance with this DSRS section.

4. Compliance with GDC 31 requires that, under the stress of operation, maintenance, testing, and postulated accidents, the reactor coolant pressure boundary shall be designed with sufficient margin to ensure that (1) the boundary behaves in a nonbrittle manner, and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other boundary material variables under a full-range of conditions. The design will also address such issues as the uncertainties of determining material properties; the effects of irradiation on material properties; residual, steady state, and transient stresses; and the sizes of flaws.

   GDC 31 is applicable to this section because the reviewer evaluates steam system piping failures, both inside and outside containment that could cause transient conditions with a potentially harmful effect on the reactor coolant pressure boundary. A steam system piping break can result in a rapid decrease in reactor coolant temperature and steam generator pressure, placing undue stress on the reactor coolant pressure boundary. The amount of stress to the reactor coolant pressure boundary depends on the severity of the transient. The severity of the transient is assessed by the applicant in the technical submittal and is reviewed by the staff in accordance with this DSRS section.

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.

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 (EQ) of Electrical Equipment 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 reviewer determines the acceptability of the analytical models and assumptions, as follows:

A. The values of system parameters and initial core and system conditions used as input to any analytical model are reviewed by the branch that reviews reactor systems. Of particular importance is (1) the reactivity feedback and control rod worth used in the analysis, and (2) the variation of moderator temperature, void and Doppler reactivity feedback with core life. The reviewer will evaluate the justification supplied by the applicant to show that the core burnup yielding the minimum margins has been selected. The branch reviewing core performance reviews core-related parameters, such as DNBR correlations and the values of the reactivity parameters used in the analysis. The reviewer confirms that the amount of secondary coolant expelled from the system (for breaks outside containment) has been calculated conservatively by evaluating the methods and assumptions, by comparing these results with those of an acceptable analysis performed on another plant of similar design, or by comparing the results with staff calculations.

B. The acceptability of the methods equations, sensitivity studies, and models proposed by the applicant are evaluated.

C. Analytical models should be sufficiently detailed to simulate the reactor coolant (primary), steam generator (secondary), and auxiliary systems. The reviewer evaluates the following functional requirements:

   i. Reactor trip signal: Credit taken for any reactor trip signal is reviewed by the branch that reviews instrumentation and control to confirm that, under accident conditions, the instrumentation and control systems are capable of the assumed response.

   ii. ECC system: In the NuScale design, the ECC system does not have the capability to makeup to the reduced RCS volume due to the cooldown. Therefore, the reviewer should ensure the RCS water level does not drop below a value that would inhibit decay heat removal. This is done by the branch reviewing plant systems as to availability of the system and by the branch reviewing reactor systems as to the capability to affect an orderly shutdown. In the case of NuScale, the safety-related functions of decay heat removal are performed by the DHRS.

   iii. Decay Heat Removal System (DHRS): The credit taken for actuation of the DHRS is reviewed by the branch that evaluates instrument and control systems to verify the ability of the instrumentation and control systems to respond as assumed. Because these systems could require both automatic and manual actuation, preoperational tests should be specified to identify any necessary operator actions and to establish times required for their completion.

D. Time-related variations of the following parameters are reviewed:

   i. reactor power

   ii. heat fluxes (average and maximum)

   iii. total core reactivity

iv. RCS pressure

v. minimum DNBR

vi. coolant conditions (inlet temperature core average temperature and average exit and hot channel exit temperatures)

vii. fuel rod conditions (maximum fuel centerBline temperature, maximum clad temperature, or maximum fuel enthalpy)

viii. steam generator pressure

ix. containment pressure

x. relief and/or safety valve flow rates

xi. discharge flow rate

xii. steam line and feedwater flow rates

xiii. pressurizer and steam generator water levels

The values of the more important of these parameters for the steam line break accident (as listed in Subsection I) are compared with those predicted for other similar plants, if available, to see that they are within the range expected.

4. To the extent deemed necessary, the reviewer evaluates the effect of single active failures of systems and components that may affect the course of the accident. For new applications, LOOP should not be treated as a single active failure, as discussed under Subsection II, Assumptions 2 and 4. This phase of the review is done using the system review procedures described in the DSRS sections for Chapters 5, 6, 7, 8, and 10 of the technical submittal. The reviewer also considers single failures that may cause more than one steam generator to blow down or failure of main feedwater to isolate, thus increasing the reactivity addition to the core.

5. The reviewer confirms that a commitment has been made in the technical submittal to conduct preoperational tests for verifying that valve discharge rates and response times (including, for example, opening and closing times for turbine and main steam isolation valves, and steam generator and pressurizer relief and safety valves) have been conservatively modeled in the accident analyses. In addition, preoperational testing should include verification of reactor trip delay times, startup delay times for DHRS, and delay times for delivery of any high concentration boron solution required to bring the plant to a safe-shutdown condition, if applicable.

6. Based on the above information, the branch that reviews radiological consequences of design basis accidents evaluates the radiological consequences of the design-basis steam line break accident as described in the DSRS Section 15.0.3.

7. Upon request from the primary reviewer, other secondary reviewers will provide input for the areas of review stated in Subsection I of this DSRS section. The primary reviewer obtains and uses such input as required to ensure that this review procedure is complete.

8. 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 technical submittal meets the acceptance criteria. DCs have referred to the technical submittal as the design control document (DCD). The reviewer should also consider the appropriateness of identified COL action items. The reviewer may identify other COL action items; however, to ensure these COL action items are addressed during a COL application, they should be added to the DC technical submittal.

   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 or other NRC approvals (e.g., manufacturing license, 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 DSRS Introduction, support conclusions of the following type to be included in the staff's SER. The reviewer also states the bases for those conclusions.

The staff concludes that the consequences of postulated steam line breaks meet the relevant requirements set forth in the GDCs 13, 17, 27, 28, and 31, regarding (1) the ability to insert the control rods and to cool the core, and (2) TMI Action Plan items. This conclusion is based upon the following:

1. The applicant meets GDC 13 requirements by demonstrating that all credited instrumentation was available, and that actuations of protection systems, automatic and manual, occurred at values of monitored parameters that were within the instruments' prescribed operating ranges.

2. The applicant has met the requirements of GDCs 27 and 28 by demonstrating that the resultant fuel damage was limited such that the ability to insert control rods would be maintained and that no loss of core cooling capability resulted. The minimum DNBR experienced by any fuel rod was _________, resulting in __% of the rods experiencing cladding perforation.

3. The applicant has met the requirements of GDC 31 with respect to demonstrating the integrity of the primary system boundary to withstand postulated accidents.

4. The analyses and effects of steam line break accidents inside and outside containment, during various modes of operation with and without offsite power (as required by GDC 17), have been reviewed and were evaluated using a mathematical model that has been previously reviewed and found acceptable by the staff.

5. The parameters used as input to this model were reviewed and found to be suitably conservative.

6. The applicant has met the requirements of 10 CFR 50.34(f)(1)(ii) and 10 CFR 50.34(f)(2)(xii) with respect to demonstrating the adequacy of the design of the

DHRS or other qualified systems to remove decay heat following steam system piping failures.

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.

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*, "Domestic Licensing of Production and Utilization Facilities," Part 50, Title 10, "Energy," Appendix A, General Design Criterion (GDC) 13, "Instrumentation and Control."

2. *U.S. Code of Federal Regulations*, "Domestic Licensing of Production and Utilization Facilities," Part 50, Title 10, "Energy," Appendix A, GDC 17, "Electric Power Systems."

3. *U.S. Code of Federal Regulations*, "Domestic Licensing of Production and Utilization Facilities," Part 50, Title 10, "Energy," Appendix A, GDC 27, "Combined Reactivity Control Systems Capability."

4. *U.S. Code of Federal Regulations*, "Domestic Licensing of Production and Utilization Facilities," Part 50, Title 10, "Energy," Appendix A, GDC 28, "Reactivity Limits."

5. *U.S. Code of Federal Regulations*, "Domestic Licensing of Production and Utilization Facilities," Part 50, Title 10, "Energy," Appendix A, GDC 31, "Fracture Prevention of Reactor Coolant Pressure Boundary."

6. U.S. Nuclear Regulatory Commission, "Protection against Postulated Piping Failures in Fluid Systems outside Containment," BTP 3-3, Revision 3, March 2007, Agencywide Documents Access and Management System (ADAMS) Accession No. ML070800027.

7. U.S. Nuclear Regulatory Commission, "Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment," BTP 3-4, Revision 2, March 2007, ADAMS Accession No. ML070800008.

8. 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 (ML102510405)