Draft for Comment

U.S. NUCLEAR REGULATORY COMMISSION

DESIGN-SPECIFIC REVIEW STANDARD FOR NuScale SMR DESIGN

15.2.1-15.2.5 LOSS OF EXTERNAL LOAD; TURBINE TRIP; LOSS OF CONDENSER VACUUM; CLOSURE OF MAIN STEAM ISOLATION VALVE; AND STEAM PRESSURE REGULATOR FAILURE (CLOSED)

REVIEW RESPONSIBILITIES

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

Secondary - None

I. AREAS OF REVIEW

A number of initiating events that occur with moderate frequency result in unplanned decreases in heat removal by the secondary system. Each event covered in this Design-Specific Review Standard (DSRS) section should be addressed in individual sections of the applicant's technical submittal) as specified in Regulatory Guide (RG) 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants and Regulatory Guide 1.206", "Combined License Applications for Nuclear Power Plants (LWR Edition)."

The DHRS consists of two natural convection-driven heat exchanger trains. Each train comprises a loop that includes a steam generator and a decay heat removal heat exchanger (DHR HX). Primary system water circulates by natural convection through the steam generators within the reactor vessel and transfers energy to the secondary side of the steam generator. The secondary coolant circulates through the heat transfer loop that includes the steam generator and DHR HX by natural convection, and transfers energy to the reactor pool through the DHR HX.

The specific areas of review are as follows:

1. Loss of External Load: In a loss of external load event, an electrical disturbance causes loss of a significant portion of the generator load. Following the loss of load without operation of either turbine bypass or the main steam pressure relief (PORVs), atmospheric dump, or safety relief valves, there is a sudden reduction in steam flow causing the pressure and temperature in the shell side of the steam generator to increase. The latter effect, in turn, results in an increase of reactor coolant temperature, a decrease in coolant density, an increase of water volume in the pressurizer, and an increase in reactor coolant pressure. The reactor trips, and the decay heat removal system (DHRS) is initiated. Decay heat is transferred by natural circulation to the reactor pool.

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2. <u>Turbine Trip</u>: In a turbine trip event, a malfunction of a turbine or reactor system causes the turbine to trip off the line by abruptly stopping steam flow to the turbine. Without operation of either the turbine bypass or the main steam pressure relief (PORVs), atmospheric dump, or safety relief valves, there is a sudden reduction in steam flow causing the pressure and temperature in the tube side of the steam generator to increase. The latter effect, in turn, results in an increase of reactor coolant temperature, a decrease in coolant density, an increase of water volume in the pressurizer, and an increase in reactor coolant pressure. The reactor trips and the DHRS is initiated. Decay heat would be transferred to the reactor pool by natural circulation. This event may be different from the loss of external load conditions as a result of differences in the steam flow reduction time scale.

3. <u>Loss of Condenser Vacuum</u>: A loss of condenser vacuum event is a malfunction that can result in a turbine trip; thus, the remarks in paragraph 2 apply to this event. In addition, due to system interaction, the loss of condenser vacuum event also causes the condensate and feedwater pumps to trip due to low suction pressure. The corresponding peak pressure in the primary and secondary systems requires separate analysis because the initial conditions that lead to peak pressure are different for the primary and secondary systems.

4. <u>Main Steam Isolation Valve (MSIV) Closure</u>: The effect of MSIV closure is limited steam flow to the turbine. The results are similar to those addressed in paragraph 1.

5. <u>Steam Pressure Regulator Failure</u>: Steam pressure regulator failure in a closed position yields a transient similar to those previously addressed. Generally, the rate of change of system parameters is slower for a steam pressure regulator failure, and a less severe transient results; thus, the steam pressure regulator failure response plays a significant role in the severity of this transient.

6. Review of these five described transients includes the sequence of events, the analytical models, the values of parameters in the analytical models, and the predicted consequences of the transients.

    A. Review is performed to ensure that the sequence of events described in the applicant's technical submittal analysis is for reactor systems and instrumentation and control. The reactor systems review concentrates on the assumptions for the reactor protection system, the engineered safety systems, and required operator actions to secure and maintain the reactor in a safe condition.

    B. The reactor systems review includes the analytical methods for whether all mathematical models and computer codes have been reviewed and accepted by the staff. If a referenced analytical method or code has not been reviewed, a generic evaluation of the new analytical model or code is performed.

    C. The results of the analyses are reviewed for whether predicted values of pertinent system parameters are within expected ranges for the type and class of reactor under review. The predicted results of the transient analyses then are reviewed for whether the consequences meet the acceptance criteria of subsection II of this DSRS section.

D. The reactor systems review includes the values of all parameters in the analytical models, including the initial conditions of the core and systems. In addition, a review is performed for core physics, fuel design, and core thermal-hydraulics data in the applicant's technical submittal analysis as part of the review of sections corresponding to DSRS Sections 4.2 through 4.4. Finally, the reactor systems review includes applicant's technical submittal Section 5.2.2 for adequacy of the overpressure protection of the reactor coolant pressure boundary (RCPB).

7. Combined Operating License (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. 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. Aspects of the transient 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 Sections 7.2 and 7.3.

4. Potential bypass modes and the possibility of manual control by the operator are reviewed under DSRS Sections7.0 through 7.2.

5. Technical specifications and short-term availability are reviewed under DSRS Section 16.0.

6. The determination of the safety-related and risk significance of SSCs relied upon to meet required functions during the accidents are based on the review of the probabilistic risk analysis under Standard Review Plan (SRP) Chapter 19.

II. ACCEPTANCE CRITERIA

Requirements

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

1. General Design Criterion (GDC) 10, Reactor Design.,

2. GDC 13, Instrumentation and Control.

3. GDC 15, Reactor Coolant System Design.

4. GDC 1'7, Electric Power Systems.

5. GDC 26, Reactivity Control System Redundancy and Capability.

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. The basic objectives of the review of the initiating events listed in subsection I of this DSRS section:

   A. To identify which moderate-frequency event that results in an unplanned decrease in secondary system heat removal is the most limiting, in particular as to primary pressure, secondary pressure, and long-term decay heat removal.

   B. To verify whether the predicted plant response for the most limiting event satisfies the specific criteria for fuel damage and system pressure.

   C. To verify whether the plant protection systems setpoints assumed in the transients analyses are selected with adequate allowance for measurement inaccuracies as delineated in RG 1.105.

   D. To verify whether the event evaluation considers single failures, operator errors, and performance of nonsafety-related systems consistent with the RG 1.206 regulatory guidelines.

2. With the ANS standards as guidance, specific criteria meet the relevant requirements of GDCs 10, 13, 15, 17, and 26 for events of moderate frequency.

   A. Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.

B. Fuel cladding integrity must be maintained by the minimum departure from nucleate boiling ratio (DNBR) remaining above the 95/95 DNBR limit based on acceptable correlations (see applicant's technical submittal Section 4.4) and by satisfaction of any other SAFDL applicable to the particular reactor design.

C. An incident of moderate frequency should not generate an aggravated plant condition without other faults occurring independently.

D. The requirements in RG 1.105, "Instrument Spans and Setpoints," are used for their impact on the plant response to the type of AOOs addressed in this DSRS section.

E. The most limiting plant system single failure, as defined in "Definitions and Explanations," 10 CFR Part 50, Appendix A, must be assumed in the analysis according to the guidance of RG 1.53 and GDC 17.

F. Performance of nonsafety-related systems during transients and accidents and single failures of active and passive systems (especially as to the performance of check valves in passive systems) must be evaluated and verified according to the guidance of SECY 77-439, SECY 94-084, and RG 1.206

3. The applicant should analyze these events using an acceptable analytical model. Any other analytical method proposed by the applicant is evaluated by the staff for acceptability. For new generic methods, the reviewer requests an evaluation by the appropriate organization for reactor systems.

The values of the parameters in the analytical model should be suitably conservative. The following values are acceptable:

A. The reactor is initially at 102 percent of the rated (licensed) core thermal power (to account for a 2 percent power measurement uncertainty unless a lower number can be justified through measurement uncertainty methodology and evaluation or unless the uncertainty otherwise is accounted for (see applicant's technical submittal Section 4.4)), and primary loop flow is at the nominal design flow less the flow measurement uncertainty.

B. Conservative scram characteristics are assumed (maximum time delay with the most reactive rod held out of the core) unless (i) a different conservatism factor can be justified through the uncertainty methodology and evaluation or (ii) the uncertainty is otherwise accounted for (see applicant's technical submittal Section 4.4).

C. The core burn-up is 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.

D. Mitigating systems should be assumed to be actuated in the analyses at setpoints with allowance for instrument uncertainty in accordance with Regulatory Guide 1.105.

Programmatic Requirements: The NRC regulations require that each operating license contain a technical specification (TS) that define "…the limits, operating conditions, and other requirements imposed upon facility operation for the protection of public health and safety…" The licensee's analysis of DSRS 15.2.1-15.2.5 must be consistent with the information presented in the licensee'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. GDC 10 requires design of reactor core and its coolant, control, and protection systems with appropriate margin so SAFDLs are not exceeded during any conditions of normal operation, including the effects of AOOs.

   GDC 10 applies to this section because the reviewer evaluates the consequences of AOOs that could decrease heat removal by the secondary system and result in the fuel cladding thermal design criteria to be exceeded. RG 1.105 provides guidance for keeping instrument setpoints within technical specification limits.

   GDC 10 requirements provide assurance that SAFDLs are not exceeded for initiating events that decrease heat removal by the secondary system.

2. GDC 13 requires the provision of instrumentation that is capable of monitoring variables and systems over their anticipated ranges to assure adequate safety, and of controls that can maintain these variables and systems within prescribed operating ranges.

   GDC 13 applies to this section because the reviewer evaluates the sequences 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 instruments indications.

3. GDC 15 requires design of the reactor coolant system and its auxiliary, control, and protection systems with sufficient margin so RCPB design conditions are not exceeded during any conditions of normal operation, including AOOs.

   GDC 15 applies to this section because the reviewer evaluates the consequences of AOOs that could decrease heat removal by the secondary system and lead to an increase in the reactor coolant temperature and pressure.

   GDC 15 requirements provide assurance that RCPB design conditions are not exceeded for initiating events that decrease heat removal by the secondary system.

4. GDC 17 requires onsite and offsite electrical power systems for safety-related SSCs to perform intended functions. Each power system (assuming the other system is not functioning) must provide sufficient capacity and capability so SAFDLs and RCPB design conditions are not exceeded in AOOs.

   GDC 17 applies to this DSRS section because it governs review of the analysis of abnormal operating occurrences to which it must be applied.

   GDC 17 requirements provide assurance that SAFDLs and RCPB design conditions are not exceeded in initiating events that decrease heat removal by the secondary system, concurrent with a loss of offsite power (LOOP).

5. GDC 26 requires two independent reactivity control systems with different design principles to control reactivity changes so acceptable fuel design limits are not exceeded.

   GDC 26 applies to this section because the reviewer evaluates the consequences of AOOs that could decrease heat removal by the secondary system and lead to reactivity changes within the core causing the fuel cladding thermal design criteria to be exceeded.

   GDC 26 requires reactivity control systems to control reactivity changes reliably with appropriate margin for malfunctions (*i.e.*, stuck control rods) so that under conditions of normal operation, including AOOs, SAFDLs are not exceeded. Where applicable, the reviewer examines these margins for whether thermal criteria are satisfied.

   GDC 26 requirements provide assurance that SAFDLs are not exceeded, ensuring an appropriate margin for malfunctions of the reactivity control system.

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.

The procedures are used for the design certification (DC) application review, the construction permit (CP), operating license (OL), and COL reviews. During below the CP review, the values of system parameters and setpoints in the analysis are preliminary and subject to change. At the OL or COL review stage, final values should be in the analysis, and the reviewer should compare these to the limiting safety system settings in the proposed technical specifications.

1. The applicant's technical submittal description of these transients is reviewed for the occurrences leading to the initiating event. The sequence of events from initiation until a stabilized condition is reached is reviewed for:

   A. The extent to which normally operating plant instrumentation and controls are assumed to function.
   B. The extent to which plant and reactor protection systems are required to function.
   C. The extent to which credit is taken for the functioning of normally operating plant systems.
   D. The extent to which operation of engineered safety systems is required.
   E. The extent to which operator actions are required.
   F. Appropriate margin for malfunctions (*e.g.*, stuck rods).
   G. Appropriate accounting for instrumentation uncertainties of system and operating parameters.

2. If the applicant's technical submittal states that one of these transients is not as limiting as other similar transients, the reviewer evaluates the applicant's justification. The Applicant's technical submittal must present a quantitative analysis of the most limiting reduction-of-heat-removal transient. For this transient, the reactor systems reviewer, in consultation with the instrumentation and controls reviewer, reviews the timing of the initiation of protection, engineered safety, and other systems needed to limit the consequences of the transient adequately to an acceptable level. The reactor systems reviewer compares the predicted variation of system parameters with various trip and system initiation setpoints. The instrumentation and controls reviewer consults on automatic initiation, actuation delays, possible bypass modes, interlocks, and the feasibility of manual operation if the applicant's technical submittal states that operator action is needed or expected.

3. To the extent deemed necessary, the reviewer evaluates the effect of single active system or component failures that may affect the course of the transient. For new applications, LOOP should not be considered a single failure; each of the reduction-of-heat-removal transients should be analyzed with and without a LOOP in combination with a single active failure. This phase of the review uses the system review procedures described in the DSRSs for applicant's technical submittal Chapters 5, 6, 7, and 8.

4. The applicant's mathematical models to evaluate core performance and to predict system pressure in the reactor coolant system and main steam line are reviewed by the organization responsible for reactor systems for whether these models have been reviewed and accepted by the staff. If not, the organization responsible for reactor systems initiates a generic review of the applicant's proposed model.

5. The values of system parameters and initial core and system conditions as input to the model are reviewed by the organization responsible for reactor systems. Of particular importance are (A) the reactivity feedback and control rod worths in the applicant's analysis and (B) the variations of moderator temperature, void, and Doppler coefficients of reactivity with core life. The reviewer evaluates the applicant's justification showing that the core burn-up selected yields the minimum safety margins.

6. The results of the analysis are reviewed and compared to the acceptance criteria of subsection II of this DSRS section for fuel integrity, the possibility of the event becoming more serious, and the maximum pressure in the reactor coolant and main steam systems. The following parameters are reviewed:

   A. reactor power;
   B. heat fluxes (average and maximum);
   C. reactor coolant system pressure;
   D. minimum DNBR (PWR) or CPR (BWR);
   E. core and recirculation loop coolant flow rates (BWR);
   F. coolant conditions (inlet temperature, core average temperature (PWR), core average steam volume fraction (BWR), average exit and hot channel exit temperatures, and steam fractions);
   G. steam line pressure;
   H. containment and suppression pool (if applicable) pressures and temperatures;
   I. maximum pressurizer water volume;
   J. pressure safety and relief valve flow rates; and
   K. flow rate from the reactor coolant system to the containment system (if applicable).

   The more important parameters for the limiting transient are compared to those predicted for similar plants for whether they are within expected range.

7. 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 applicant’s technical submittal meets the acceptance criteria. DCs have referred to the applicant’s technical submittal as the design control document (DCD). The reviewer should also consider the appropriateness of identified COL action items. The reviewer may identify additional COL action items; however, to ensure these COL action items are addressed during a COL application, they should be added to the DC applicant’s 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 (ESP) 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 review and calculations (if applicable) 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 plant design is acceptable as to transients resulting in unplanned decreases in heat removal by the secondary system, transients expected with moderate frequency, and transients where the predicted response meets the requirements of GDCs 10, 13, 15, 17, and 26. This conclusion is based on the following findings:

1. The applicant meets the requirements of GDCs 10 and 26 by demonstrating that SAFDLs are not exceeded for this event. The applicant also meets GDC 15 requirements by preventing plant transients from resulting in unplanned decreases in heat removal by the secondary system and demonstrating reactor coolant pressure limits not exceeded by these events and resultant leakage within acceptable limits.

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

3. The transient initiating events that might occur with moderate frequency are:

   A. turbine trip,
   B. loss of external load,
   C. steam pressure regulator malfunctions,
   D. main steam isolation valve closure,
   E. loss of condenser vacuum,
   F. loss of nonemergency AC power to the station auxiliaries,
   G. loss of normal feedwater flow.[1]

4. In a review of the transients that could result from these postulated events, it was found that the most limiting in regard to core thermal margins and pressure within the reactor coolant and main steam systems was the ____________ transient. This transient was evaluated by the applicant using a mathematical model that had been previously reviewed and found to be acceptable by the staff. The parameters used as input to this model were reviewed and found to be suitably conservative and in accordance with the recommendation of RG 1.105. The results of the analysis of the transient showed that cladding integrity was maintained by ensuring that the minimum departure from nucleate boiling ratio (or minimum critical power ratio for a BWR) did not decrease below _________ and that the maximum pressure within the reactor coolant and main steam systems did not exceed 110% of their design pressures.

5. The applicant meets the requirements of GDCs 17 and 26 by demonstrating that SAFDLs are not exceeded for this event. In addition, the applicant meets GDC 15 requirements by demonstrating that the reactor coolant pressure limits are exceeded by this event and that resultant leakage is within acceptable limits.

6. The applicant meets the positions of RG 1.53, SECY 77-439, SECY 94-084 and RG 1.206 on the single-failure criterion and RG 1.105 on instrument actuations of safety-related systems and components.

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[1] The SER should present one statement for moderate frequency transients involving unplanned decrease in heat removal by the secondary system; thus, the results of reviews under DSRS Sections 15.2.6 and 15.2.7 are included in this statement.

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 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. 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."

2. 10 CFR Part 50, Appendix A, GDC 10, "Reactor Design."

3. 10 CFR Part 50, Appendix A, GDC 13, "Instrumentation and Control."

4. 10 CFR Part 50, Appendix A, GDC 15, "Reactor Coolant System Design."

5. 10 CFR Part 50, Appendix A, GDC 17, "Electric Power Systems."

6. 10 CFR Part 50, Appendix A, GDC 26, "Reactivity Control System Redundancy and Capability."

7. 10 CFR Part 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants."

8. RG 1.53, "Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems."

9. RG 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants."

10. RG 1.105, "Instrument Spans and Setpoints."

11. NUREG-0718, "Licensing Requirements for Pending Applications for Construction Permits and Manufacturing Licenses."

12. NUREG-0737, "Clarification of TMI Action Plan Requirements."

13. SECY-77-439, "Single Failure Criterion."

14. SECY-94-084, "Policy and Technical Issues Associated With the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs."

15. SECY-94-132, "Policy and Technical Issues Associated With the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs."

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

17. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," Article NB-7000, "Protection Against Overpressure," American Society of Mechanical Engineers.

18. ANSI/ANS-51.1-1983, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," (replaced ANSI N18.2-1974; reaffirmed 1988; withdrawn 1998).