Draft for Comment

U.S. NUCLEAR REGULATORY COMMISSION

DESIGN-SPECIFIC REVIEW STANDARD FOR NuScale SMR DESIGN

15.2.8 FEEDWATER SYSTEM PIPE BREAK INSIDE AND OUTSIDE CONTAINMENT

REVIEW RESPONSIBILITIES

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

Secondary - None

I. AREAS OF REVIEW

The steam and water release from a postulated feedwater line break results in a loss of secondary coolant which may result in a reactor system cool-down (by excessive energy discharge through the break) or a reactor system heat-up (from the loss of reactor system heat sink). A major feedwater line rupture is defined as a feedwater line break large enough to prevent the addition of sufficient feedwater to the steam generators to maintain shell side fluid inventory in the steam generators. If the break is postulated in the feedwater line between the isolation valves and the steam generator, fluid from the steam generator also is discharged from the break. (A break upstream of the feedwater isolation valves would affect the reactor system only as a loss of feedwater. This case is covered by DSRS Section 15.2.7, ”Loss of Normal Feedwater Flow.”)

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.

If a feedwater line rupture causes the water in the steam generator to be discharged through the break, the water will not be available for decay heat removal after reactor scram. The break location and size may prevent addition of any feedwater to the affected steam generator. The NuScale passive decay heat removal system (DHRS) is activated to remove decay heat to prevent over-pressurization of the reactor system.

The specific areas of review are as follow:

1. Evaluation of the applicant's postulated initial core and reactor conditions pertinent to the feedwater line break.

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The results of the analyses are reviewed for whether the values of pertinent system parameters, addressed in subsection II of this Design-Specific Review Standard (DSRS) section, are within expected ranges. The parameters of importance for these transients include:

A. reactor coolant system (RCS) pressure,
B. steam generator pressure,
C. fluid temperatures,
D. fuel and clad temperatures,
E. break discharge flow rate,
F. steamline and feedwater flow rates,
G. safety and relief valve flow rates,
H. pressurizer and steam generator water levels,
I. mass and energy transfer within the containment (for breaks inside containment),
J. reactor power,
K. total core reactivity,
L. hot and average channel heat flux, and
M. minimum departure from nucleate boiling ratio (DNBR).

2. Methods of thermal and hydraulic analysis, the postulated sequence of events, including analyses to determine the time of reactor trip and time delays prior and subsequent to initiation of reactor protection system (RPS) actions.

   The analytical thermal/hydraulic methods are reviewed for whether the mathematical modeling and computer codes have been reviewed and accepted by the staff. If a referenced analytical method has not been reviewed, the reviewer requests an evaluation of the new analytical model. The parameter values in the analytical model, the initial conditions of the core, and all nuclear design parameters are reviewed. This review includes:

   A. power level,
   B. power distribution,
   C. Doppler reactivity feedback,
   D. moderator temperature reactivity feedback,
   E. void reactivity feedback,
   F. reactor kinetics,
   G. departure from nucleate boiling correlations, and H. control rod worth.

3. The response of the reactor coolant and auxiliary systems, the functional and operational characteristics of RPS effects on the sequence of events, and all operator actions required to secure and maintain the reactor in a safe shutdown condition.

   The sequence of events described in the applicant's safety analysis report (SAR) is reviewed for the performance of the RPS, the engineered safety systems, and operator actions to secure and maintain the reactor in a safe condition.

4. The DHRS is reviewed for whether the natural circulation flow is acceptable for transient control following a feedwater line break.

5. 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 Design Certification (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. Effects of blow-down loads, including jet propulsion piping and component supports and the design bases for safety and relief valves are reviewed under Standard Review Plan (SRP) Sections 3.6.2 and 3.9.1 through 3.9.3. Design bases for safety and relief valves are also reviewed under SRP Section 3.9.3.

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 SAR analysis are reviewed under DSRS Sections 4.2, 4.3, and 4.4.

5. 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.

6. The response of the containment to feedwater line ruptures as 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 6.2.1.3.

7. Aspects of the sequence described in the SAR 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 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.

8. The DHRS is reviewed to verify its ability to function following a steam line break given a single active component failure with either onsite or offsite power under SRP Section 10.4.9.

9. Fission product release assumptions for determining any offsite releases are evaluated and radiological consequences from a feedwater pipe break are verified as within acceptable limits are reviewed under DSRS 15.6.5.

10. 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 SRP Chapter 19.

II. ACCEPTANCE CRITERIA

Requirements

The basic objective of the review of feedwater system pipe break events is to confirm that the reactor primary system is maintained in a safe status for break sizes up to and including a break equivalent in area to the double-ended rupture of the largest feedwater line.

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

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

2. GDC 17, Electric Power Systems.

3. GDCs 27, Combined Reactivity Control Systems Capability. and 28 *Reactivity Limits*

4. GDC 31, Fracture Prevention of Reactor Coolant Pressure Boundary.

5. GDC 35, Emergency Core Cooling.

6. 10 CFR Part 100, as to calculated doses at the site boundary.

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. Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design pressures (American Society of Mechanical Engineers

(ASME) Boiler and Pressure Vessel Code, Section III) for low-probability events and below 120 percent for very low-probability events like double-ended guillotine breaks.

2. The potential for core damage is evaluated for an acceptable minimum DNBR remaining above the 95/95 DNBR limit for pressurized-water reactors (PWRs) identified in [insert document title here] 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 not meeting these criteria unless, from an acceptable fuel damage model (see DSRS Section 4.2) including the potential adverse effects of hydraulic instabilities, fewer failures can be shown to occur. Any fuel damage calculated to occur must be of sufficiently limited extent that the core remains in place and intact with no loss of core cooling capability.

3. Calculated doses at the site boundary from any activity release must be a small fraction of the 10 CFR Part 100 guidelines.

4. The DHRS must be safety grade and automatically initiated when required.

5. Certain assumptions should be in the analysis of important parameters that describe initial plant conditions and postulated system failures:

   A. The power level assumed and number of steam generator-feedwater loops operating at the initiation of the transient should correspond to the operating condition which maximizes accident consequences. These assumed initial conditions vary with the particular nuclear steam supply system and sensitivity studies are required to determine the most conservative combination of power level and plant operating mode. These sensitivity studies may be presented in a generic report as references if applicable.

   B. The assumptions as to whether offsite power is lost and the time of loss should be conservative. Offsite power may be lost simultaneously with the pipe break, the loss may occur during the accident, or offsite power may not be lost. A study should determine the most conservative assumption appropriate to the plant design reviewed. The study should take account of the effects that loss of offsite power (LOOP) has on main feedwater pump trips and on the initiation of the DHRS and the consequent modification of the sequence of events.

   C. The effects (pipe whip, jet impingement, reaction forces, temperature, humidity, etc.) of the postulated feedwater line breaks on other systems should be considered consistently with the intent of Branch Technical Positions (BTP) 3-3 and BTP 3-4.

   D. The worst single active component failure should be assumed to occur in the systems required to control the transient. For new applications, LOOP should not be considered a single failure; feedwater pipe breaks should be analyzed with and without LOOP, as in assumption B, in combination with a single, active failure. (This position is based upon interpretation of GDC 17 as documented in the FSER for the ABB-CE System 80+ DC.)

E. The maximum rod worth should be assumed to be held in the fully withdrawn position per GDC 25. An appropriate rod reactivity worth versus rod position curve should be assumed.

F. 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.

G. The initial core flow assumed for the analysis of the feedwater line rupture accident should be chosen conservatively. If the minimum core flow allowed by the technical specifications is assumed, the minimum DNBR margin is the result for a feedwater line rupture inside containment; however, this assumption may not be the most conservative. For example, maximum initial core flow increases RCS cool-down and depressurization, decreases shutdown margin, and increases the possibility that the core will become critical and return to power. As it is not clear which initial core flow is most conservative, the applicant's assumption should be justified by appropriate sensitivity studies.

H. During the initial 10 minutes of the transient, if credit for operator action is required, an assessment for the limiting consequence must account for operator delay and/or error.

Programmatic Requirements: The NRC regulations require that each operating license contain a technical specification (TS) that defines "…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 15.2.8 must be consistent with the information presented in the licensee's TS.

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. 10 CFR Part 100 specifies how the exclusion area, low population zone, and population center distance should be determined. Further, 10 CFR Part 100 radiation exposure criteria provide reference values for the site suitability determination based on postulated fission product releases from accidental events.

   10 CFR Part 100 applies to this section because i describes the methodology for calculating radiation exposures at the site boundary for postulated accidents or events like loss of an RCP. For transients with moderate frequencies of occurrence, the calculated doses at the site boundary from any release of radioactive material must be a small fraction, less than 10 percent, of the 10 CFR Part 100 guidelines. For purposes of this review, consideration of the radiological consequences of any feedwater system pipe break must include the containment, confinement, and filtering systems. The applicant's source terms and methodologies as to gap release fractions, iodine chemical form, and fission product release timing should reflect NRC-approved source terms and methodologies.

2. GDC 13 requires the provision of instrumentation to monitorvariables and systems over their anticipated ranges to assure adequate safety, and of controls to 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 instrument indications.

3. GDC 17 requires onsite and an offsite electric power systems to permit functioning of SSCs important to safety. The safety function for each power system (assuming the other system is not functioning) is to provide sufficient capacity and capability to assure that (A) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences (AOOs) and (B) the core is cooled and containment and other vital functions are maintained in postulated accidents.

   GDC 17 applies because review under this section covers feedwater system pipe breaks, which can be classed as AOOs or accidents, depending upon severity.

4. GDC 27 requires reactivity control systems designed with a combined in conjunction with poison added by the emergency core cooling system, to control reactivity changes reliably to maintain core cooling capability under postulated accident conditions with appropriate margin for stuck rods.

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

   GDCs 27 and 28 apply because this DSRS section is for the review of feedwater system pipe breaks inside and outside containment that can result in transient conditions affecting reactor coolant temperature and pressure with consequent changes in core reactivity. The SAR analyses of these transients must demonstrate that reactivity, pressure, and temperature changes will not be severe enough for an unacceptable impact on the reactor coolant pressure boundary or on core cooling capability. The analyses must be reviewed by the staff independently in accordance with this DSRS section.

5. GDC 31 requires reactor pressure boundary design with sufficient margin to ensure that, when stressed under operation, maintenance, test, and postulated accident conditions, the boundary behaves in a nonbrittle manner and the probability of rapidly propagating fracture is minimized. The design must reflect consideration of service temperatures and other conditions of the boundary material under operation, maintenance, test, and

postulated accident conditions and the uncertainties in determining material properties; effects of irradiation on material properties; residual, steady state, and transient stresses; and flaw sizes.

GDC 31 applies because this DSRS section is for the review of feedwater system pipe breaks inside and outside containment that could result in transient reactor coolant temperature and pressure conditions that could affect the reactor coolant pressure boundary adversely. A feedwater system pipe break could result in either an RCS cool-down by excessive energy discharge through the break or an RCS heat-up by reduced feedwater flow to the steam generator. Heat-up of the reactor coolant by reduced feedwater flow to the steam generator and by the subsequent addition of decay heat could result in undue stress on the RCS pressure boundary. The amount of stress to which the reactor coolant pressure boundary is subjected depends upon AOO severity, which is assessed in the SAR and reviewed by the staff in accordance with this DSRS section.

6. GDC 35 requires a system for abundant emergency core cooling. The system safety function is to transfer heat from the reactor core following any loss of reactor coolant at a rate to prevent fuel and clad damage that could interfere with continued effective core cooling and limit fuel clad metal-water reaction to negligible amounts.

   GDC 35 applies because this DSRS section is for the review of feedwater system pipe breaks both inside and outside containment that could result in transient reactor coolant temperature conditions that could challenge the DHRS. A feedwater system pipe break could result in either an RCS cool-down by excessive energy discharge through the break or an RCS heat-up by reduced feedwater flow to the affected steam generator. Heat-up of the reactor coolant by reduced feedwater flow to the affected steam generator and by the subsequent addition of decay heat could initiate DHRS reduction of the core coolant temperature to an acceptable level to prevent fuel and clad damage that could interfere with continued effective core cooling and limit fuel clad metal-water reaction to negligible amounts. The severity of this AOO is assessed in the SAR and reviewed by the staff in accordance with this DSRS section.

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. The procedures are used during reviews of construction permit, operating license, and COL applications. During the construction permit review the values of system parameters and setpoints in the analysis are preliminary in nature and subject to change. At the operating license 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.

4.. The values of system parameters and initial core and system conditions as input to the model are reviewed and compared to the initial conditions listed in subsection II of this DSRS section. Of particular importance are the reactivity feedbacks and control rod worths in the applicant's analysis and the variation of moderator temperature, void, and Doppler reactivity feedbacks with core life. The applicant's justification for selection of the core burn-up yielding the minimum margins is evaluated. Reactivity parameter values in the applicant's analysis also are reviewed.

5. Analytical models should be of sufficient detail to simulate the reactor coolant (primary), steam generator (secondary), and auxiliary systems. The applicant's equations, sensitivity studies, proposed models, and justification for methods as conservative compared to appropriate test data are reviewed. The pressurizer is of particular importance in the modeling of the over-pressure transient, which is the likely result of a large feedwater line break. Assumptions for pressurizer spray performance if credited should be reviewed as well as heat transfer by condensation within the pressurizer steam space. Test data examples which might be useful in validation of pressurizer models are in "The Pressure Response of a PWR Pressurizer During an Insure Transient," Transactions of the American Nuclear Society, 1983 Annual Meeting, Detroit, MI, June 12-16, 1983.

6. Credit taken for a reactor trip signal or for ESS actuation should be reviewed for the ability of the instrumentation and control systems to respond as assumed under accident conditions.

7. The DHRS ability to supply adequate feedwater flow to the unaffected steam generators during the accident and subsequent shutdown is evaluated as to availability and capability to affect an orderly shutdown. As DHRS designs are diverse and may require both automatic and manual actuation, pre-operational tests should be specified for any necessary operator actions and for the maximum times for their completion.

   To the extent necessary, the reviewer evaluates the effect of system and component single, active failures that may alter the course of the accident. For new applications, the LOOP is not a single, active failure but an addition to a single, active failure as addressed in subsection II.6.D of this DSRS section. This phase of the review uses the system review procedures described in the DSRS sections for SAR Chapters 5, 6, 7, 8, and 10. During the transient the variations with time of parameter listed in Sections 15.X.X.3(C) and 15.X.X.4(C) of the Standard Format, Regulatory Guide 1.70, are reviewed. The more important of these parameters for the feedwater line break accident (as listed in subsection I of this DSRS section) are compared to those predicted for other similar plants for whether they are within the expected range.

8. The reviewer confirms that the amount of secondary coolant expelled from the system is calculated conservatively by evaluation of the applicant's methods and assumptions, by comparison with an acceptable analysis on another plant of similar design, or by comparison with staff calculations.

The reviewer confirms an SAR commitment to conduct pre-operational tests to verify that valve discharge rates and response times (*e.g.*, opening and closing times (delay times) for main feedwater, the DHRS, turbine and main steam isolation, and steam generator, pressurizer relief, and safety valves) are modeled conservatively in the accident analyses. In addition, pre-operational testing should include verification of reactor trip delay times, startup delay times for DHRS actuation, safety injection signal delay time, and delay times for delivery of any high-concentration boron injection required to bring the plant to a safe shutdown condition.

9. Using the information developed in the review, the reviewer evaluates the radiological consequences of the design-basis feedwater line break. This evaluation is based on a qualitative comparison to the results of the design-basis steam line break or on a detailed analysis using the approach described in the DSRS Section 15.0.3.

10. 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 applicant''s analysis of consequences of postulated feedwater line breaks meets the requirements of GDCs 13, 17, 27, 28, 31, and 35 for ability to insert control rods and ability to cool the core, 10 CFR Part 100 guidelines for radiological doses at the site boundary, and applicable Three Mile Island Action Plan Items. This conclusion is based upon the following findings:

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 meets GDCs 27 and 28 requirements by demonstrating minimal fuel damage, maintained ability to insert the control rod, and no loss of core cooling capability. The minimum DNBR for any fuel rod was _____ with the result of __ percent of the rods experiencing clad perforation.

3. The applicant meets GDC 31 requirements for demonstrating primary system boundary capability to withstand the postulated accident.

4. The applicant meets GDC 35 requirements for demonstrating emergency cooling system adequacy for abundant core cooling and reactivity control (via boron injection).

5. The analyses of effects of feedwater line break accidents inside and outside containment during various modes of operation with and without offsite power have been reviewed and evaluated by a mathematical model previously reviewed and found acceptable by the staff.

6. The input parameters for this model were reviewed and found suitably conservative.

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, Appendix A, "General Design Criteria for Nuclear Power Plants."

   A. GDC 13, ”Instrumentation and Control.”

   B. GDC 17, "Electric Power Systems."

   C. GDC 27, "Combined Reactivity Control System Capability."

   D. GDC 28, "Reactivity Limits."

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

   F. GDC 35, "Emergency Core Cooling."

2. 10 CFR Part 100, "Reactor Site Criteria."

3. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants."

4. Branch Technical Position 3-3, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment."

5. Branch Technical Position 3-4, "Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment."

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