Draft for Comment

U.S. NUCLEAR REGULATORY COMMISSION

DESIGN-SPECIFIC REVIEW STANDARD FOR NuScale SMR DESIGN

15.1.6 LOSS OF CONTAINMENT VACUUM

REVIEW RESPONSIBILITIES

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

Secondary - None

I. AREAS OF REVIEW

1. The NuScale containment is the steel vessel surrounding the reactor pressure vessel and during normal operation is kept under a vacuum. Multiple reactor modules, where a module includes the steel containment, reside in the reactor pool building which serves as the ultimate heat sink (UHS). The loss of containment vacuum is a NuScale-specific event which involves an unplanned increase in reactor cooling system (RCS) heat removal and is expected to occur with moderate frequency, i.e., during an anticipated operational occurrence (AOO). Excessive heat removal, i.e., a heat removal rate in excess of the heat generation rate in the core, causes a decrease in moderator temperature which increases core reactivity and can lead to a power level increase and a decrease in shutdown margin. The power level increase may lead to a reactor trip. Any unplanned power level increase may result in fuel damage or excessive reactor system pressure.

   The loss of containment vacuum due to air ingress increases heat transfer from the RCS to containment and the reactor pool. The loss of vacuum could be caused by operator action or equipment malfunctions. This moderate-frequency event causes a cooldown of the RCS and results in the addition of positive reactivity in the core due to the presence of negative moderator feedback.

   The topics covered in the primary review include: postulated initial core, RCS and containment conditions which are pertinent to the loss of containment vacuum; methods of thermal and hydraulic analysis; postulated sequence of events including time delays prior to and after protective system actuation; assumed reactions of reactor system components; functional and operational characteristics of the reactor protection system in terms of how it affects the sequence of events; and all operator actions required to secure and maintain the reactor in a safe condition.

   The results of the transient analysis are reviewed to ensure that the values of pertinent system parameters are within the ranges expected for the type and class of reactor under review. The parameters include: core flow and flow distribution, channel heat flux (average and hot), departure from nucleate boiling ratio (DNBR), thermal power, vessel and containment pressures, and reactivity.

   The staff reviews the sequence of events described in the technical submittal. The reviewer concentrates on the need for the reactor protection system, the engineered safety systems, and operator action to secure and maintain the reactor in a safe

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condition. The analytical methods are reviewed to ascertain whether mathematical modeling and computer codes have been previously reviewed and accepted by the staff. If a referenced analytical method has not been previously reviewed, the reactor systems reviewer initiates a generic evaluation of the new analytical model. In addition, the values of all the parameters used in the new analytical model, including the initial conditions of the core and system, are reviewed.

2. Combined License Action Items and Certification Requirements and Restrictions. For a design certification (DC) application, the review will also address combined license (COL) action items and requirements and restrictions (e.g., interface requirements and site parameters).

   For a COL application referencing a DC, a COL applicant must address COL action items (referred to as COL license information in certain DCs) included in the referenced DC. Additionally, a COL applicant must address requirements and restrictions (e.g., interface requirements and site parameters) included in the referenced DC.

Review Interfaces

Other design specific review standard (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. Instrumentation and control aspects of the sequence described in the technical submittal are reviewed to confirm that reactor and plant protection and safeguards controls and instrumentation systems will function as assumed in the safety analysis under DSRS Sections 7.0 through 7.2.

4. Technical specifications are reviewed under Standard Review Plan (SRP) Section 16.0.

5. The staff reviews the values of the parameters used in the analytical models relating to the reactor core for conformance to plant design and specified operating conditions; determines the acceptance criteria for fuel cladding damage limits; and reviews the core physics, fuel design, and core-thermal-hydraulics data used in the SAR analysis under DSRS Sections 4.2 through 4.4.

6. Risk classifications are reviewed under SRP Section 19.3.

7. DSRS Section 6.2 for containment initial conditions and models used to determine final conditions for a loss of vacuum.

II. ACCEPTANCE CRITERIA

Requirements

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

1. General Design Criterion 10 (GDC 10), as it relates to the reactor coolant system being designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations including anticipated operational occurrences.

2. GDC 13, as to the availability of instrumentation to monitor variables and systems over their anticipated ranges to assure adequate safety, and of appropriate controls to maintain these variables and systems within prescribed operating ranges.

3. GDC 15, as it relates to the reactor coolant system and its associated auxiliaries being designed with appropriate margin to ensure that the pressure boundary will not be breached during normal operations including anticipated operational occurrences.

4. GDC 20, as it relates the reactor protection system being designed to initiate automatically the operation of appropriate systems, including the reactivity control systems, to ensure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including anticipated operational occurrences.

5. GDC 26, as it relates to the reliable control of reactivity changes to ensure that specified acceptable fuel design limits are not exceeded, including anticipated operational occurrences. This is accomplished by ensuring that appropriate margin for malfunctions such as stuck rods are accounted for.

6. Section 20.1406 of Title 10 *Code of Federal Regulations* (10 CFR 20.1406) as it relates to the minimization to the extent practicable, of contamination of the facility and the environment, and designs and procedures to facilitate eventual decommissioning, and to minimize, to the extent practicable, the generation of radioactive waste.

<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.

The basic objectives of the review of the transients which result from an increase in heat removal are:

1. To identify which of the moderate-frequency initiating events that result in increased heat removal are the most limiting.

2. To verify that, for the most limiting initiating events, the plant responds to the transients in such a way that the criteria regarding fuel damage and system pressure are met.

The specific criteria necessary to meet the requirements of GDC 10, 15, 20, and 26 for incidents of moderate frequency are:

1. Pressure in the reactor coolant should be maintained below 110% of the design values.

2. Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit based on acceptable correlations (see DSRS Section 4.4).

3. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

4. To meet the requirements of GDC 10, 13, 15, 20, and 26 the positions of RG 1.105, "Instrument Spans and Setpoints," are used with regard to their impact on the plant response to the type of transient addressed in this DSRS section.

5. The most limiting plant systems single failure, as defined in the "Definitions and Explanations" of Appendix A to 10 CFR Part 50, shall be identified and assumed in the analysis and shall satisfy the positions of RG 1.53.

The applicant's analysis of transients caused by excessive heat removal should be performed using an acceptable analytical model, the NRC approved methodologies, and the computer codes. If the applicant proposes to use analytical methods which have not been approved, these methods are evaluated by the staff for acceptability. For new generic methods, the reviewer performs an evaluation based on DSRS section 15.0.2, "Transient and Accident Analysis methods."

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

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

2. Conservative scram characteristics are assumed (i.e., the maximum time delay is used with the most reactive rod held out of the core).

3. 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.

4. Containment conditions are such to maximize heat transfer from the RCS to containment and the reactor pool.

5. Mitigating systems should be assumed to be actuated in the analyses at setpoints with allowance for instrument inaccuracy in accordance with RG 1.105. Compliance with RG 1.105 is determined by the instrumentation and control systems.

<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 10 requires that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

GDC 10 is applicable to this section because the reviewer evaluates the consequences of anticipated operational occurrences that have the potential to exceed allowable thermal design criteria for fuel cladding integrity. These anticipated operational occurrences involve the transient increase in heat removal by the loss of containment vacuum, which in turn causes reactor power to increase in response to the resultant lowering of the temperature of the reactor coolant. RG 1.53 provides guidance with respect to the application of the single failure criterion to the design and analysis of nuclear power plant protection systems. RG 1.105 provides guidance for ensuring that instrument setpoints are initially within and remain within the technical specification limits.

Meeting the requirements of GDC 10 provides assurance that specified acceptable fuel design limits are not exceeded for the anticipated operational occurrences evaluated in this DSRS section involving excessive heat removal caused by the loss of containment vacuum.

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 sequence of events, including automatic actuations of protection systems, and manual actions, and determines whether the sequence of events is justified, based upon the expected values of the relevant monitored parameters and instrument indications.

3. Compliance with GDC 15 requires that the reactor coolant system and associated auxiliary, control, and protection systems be designed with sufficient margin to ensure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

   GDC 15 is applicable to this section because this overcooling event causes the reactor coolant system pressure to change in response to the drop in reactor coolant temperature.

   Meeting the requirements of GDC 15 provides assurance that the design conditions of the reactor coolant pressure boundary are not exceeded for the anticipated operational occurrences evaluated in this DSRS section involving excessive heat removal.

4. Compliance with GDC 20 requires that the reactor protection system be designed to initiate the operation of appropriate systems automatically, including the reactivity control systems, to ensure that specified acceptable fuel design limits are not exceeded during normal operation, including anticipated operational occurrences.

   GDC 20 is applicable to this section because the reviewer evaluates the reactor protection system that operates to shut down the reactor automatically to terminate the events (anticipated operational occurrences) analyzed in this DSRS section. The events are terminated by the reactor protection system in a timely manner such that fuel cladding integrity is maintained. For the NuScale small modular reactor (SMR), this means that the minimum value of the departure from (DNBR) reached during the transient must remain above the 95/95 DNBR limit for the applicable DNBR correlation.

   Meeting the requirements of GDC 20 provides assurance that specified acceptable fuel design limits are not exceeded by ensuring that the reactor protection system acts in a timely manner to terminate reactor operation prior to reaching a safety limit.

5. Compliance with GDC 26 requires that one of the reactivity control systems be control rods capable of reliably controlling reactivity changes to ensure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded.

   GDC 26 is applicable to this section because the reviewer evaluates the movement of control rods in response to the initiating event, and rod misalignment, including stuck rods, can produce more severe thermal-hydraulic conditions than would otherwise exist. GDC 26 requires that the thermal margin be sufficient to accommodate these conditions. DSRS Section 15.1.6 examines these margins where applicable to ensure that the thermal criteria limits are not exceeded.

   Meeting the requirements of GDC 26 provides assurance that specified acceptable fuel design limits are not exceeded by ensuring that there is appropriate margin for malfunctions of the reactivity control system, including stuck rods.

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 DC or COL 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 reactor systems branch reviews the applicant's description of the transients caused by excessive heat removal with specific attention to the occurrences that lead to the initiating event. The sequence of events from initiation until a stabilized condition is reached is reviewed to ascertain:

   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 credit taken for the functioning of normally operating plant systems;

   D. the operation of engineered safety systems that is required;

   E. the extent to which operator actions are required; and,

   F. that appropriate margin for malfunctions, such as stuck rods, is accounted for.

4. If the technical submittal states that a particular initiating event involving an increase in heat removal is not as limiting as some other similar event, the reviewer evaluates the justification presented by the applicant. The applicant is to present a quantitative analysis in the technical submittal of the increase-in-heat-removal event that is determined to be most limiting. For this event, the reactor systems reviewer, with the aid of the instrumentation and control systems reviewer, reviews the timing of the initiation of those protection, engineered safety, and other systems needed to limit the consequences of the transient 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 control systems review of Chapter 7 of the technical submittal confirms that the instrumentation and control systems design is consistent with the requirements for safety systems actions for these events.

5. To the extent deemed necessary, the reactor systems reviewer evaluates the effect of single active failures of systems and components which may affect the course of the transient. This phase of the review uses the system review procedures described in the SRP and DSRS sections for Chapters 4, 5, 6, 7, 8, and 9 of the technical submittal.

6. The mathematical models used by the applicant to evaluate core performance and to predict system pressure in the reactor coolant system and containment are reviewed by the reactor systems and the containment branch to determine if these models have been previously reviewed and found acceptable by the staff. If not, a generic review of the models is initiated. The reactor systems reviewer should work closely with the containment branch as the RCS response, including pressure, will depend on the containment heat transfer models.

7. The values of system parameters and initial core and system conditions used as input to the model are reviewed by the reactor systems. Of particular importance are the values of reactivities and control rod worths used by the applicant in this analysis, and the variations of moderator temperature, void, and Doppler reactivities with core life. The reviewer evaluates the justification provided by the applicant to show that the core burn-up selected yields the minimum margins. The branch reviewing reactor systems reviews the values of the reactivity parameters used in the applicant's analysis.

8. The results of the analysis are reviewed and compared with the acceptance criteria presented in Subsection II of this DSRS section regarding the maximum pressure in the reactor coolant and containment. Time-related variations of the following parameters are reviewed:

   A. Reactor power;

   B. Heat fluxes (average and maximum);

   C. Reactor coolant system pressure;

   D. Minimum DNBR;

   E. Coolant conditions (inlet temperature, core average temperature, average exit and hot channel exit temperatures, and steam fractions);

   F. Containment pressure; and,

   G. Steam flow from the pressurizer safeties to containment, if applicable.

   The values of the more important of these parameters, as listed in Subsection I of this DSRS section, are compared with those predicted for other similar plants, if applicable, to see that they are within the range expected.

9. 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 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 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 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 transient resulting in an unplanned increase in heat removal by the loss of containment vacuum is expected to occur with moderate frequency is acceptable and meets the requirements of GDC 10, 13, 15, 20, and 26.

1. In meeting GDC 10, 13, 15, 20, and 26 as indicated below the staff has determined that the applicant's analysis was performed using a mathematical model that has been reviewed and found acceptable by the staff. The parameters used as input to this model were reviewed and found to be suitably conservative and consistent with the plant design. In addition, the staff has further determined that the positions of RG 1.53 as related to the single failure criterion and RG 1.105 for instruments have also been satisfied.

2. The applicant has met the requirements of GDC 10, 20, and 26 with respect to demonstrating that resultant fuel integrity is maintained since the specified acceptable fuel design limits were not exceeded for this event.

3. 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.

4. The applicant has met the requirements of GDC 15 with respect to demonstrating that the reactor coolant pressure boundary limits have not been exceeded by this event and that resultant leakage will be within acceptable limits. This requirement has been met since the maximum pressure within the reactor coolant did not exceed 110 percent of the design pressures.

5. The applicant has met the requirements of GDC 20 and 26 with respect to the capability of the reactivity control system to provide adequate control of reactivity during this event while including appropriate margins for stuck rods since the specified acceptable fuel design limits were not exceeded.

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

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

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

4. 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."

5. 10 CFR Part 50, Appendix A, “GDC 10, "Reactor Design."

6. 10 CFR Part 50, Appendix A, “GDC 13, Instrumentation and Control.”

7. 10 CFR Part 50, Appendix A, “GDC 15, "Reactor Coolant System Design."

8. 10 CFR Part 50, Appendix A, “GDC 20, "Protection System Functions."

9. 10 CFR Part 50, Appendix A, “GDC 26, "Reactivity Control System Redundancy and Capability."

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

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

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

13. NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition.”