U.S. NUCLEAR REGULATORY COMMISSION

DESIGN-SPECIFIC REVIEW STANDARD FOR NuScale SMR DESIGN

15.2.7 LOSS OF NORMAL FEEDWATER FLOW

REVIEW RESPONSIBILITIES

Primary - Organization responsible for review of transient and accident analyses

Secondary - None

I. AREAS OF REVIEW

A loss of normal feedwater flow could occur from pump failures, valve malfunctions, or a loss of offsite power (LOOP). Emergency decay heat removal for the NuScale design is provided by the passive decay heat removal system (DHRS), which includes a heat exchanger that is submerged in the reactor building pool. The heat exchanger transfers heat from the primary reactor coolant to the pool, which acts as the ultimate heat sink. Loss of feedwater flow results in an increase in reactor coolant temperature and pressure which eventually requires a reactor trip to prevent fuel damage. Each event covered in this Design-Specific Review Standard (DSRS) section should be discussed in individual sections of the applicant's technical submittal, as specified in Regulatory Guide 1.70 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.

For the NuScale SMR, fission product decay heat must be transferred from the reactor coolant system following a loss of normal feedwater flow. This can be accomplished by actuation of one or several of the following systems: steam relief system, the (DHRS) and the emergency core cooling system (ECCS).

The specific areas of review are as follows:

1. The sequence of events described in the applicant's technical submittal is reviewed by both the organization responsible for reactor systems and the organization for the instrumentation and control systems. The reactor systems reviewer concentrates on the need for the reactor protection system, the DHRS, the ECCS, and operator action to secure and maintain the reactor in a safe condition.

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2. The analytical methods are reviewed by the organization responsible for reactor systems to ascertain whether the 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 requests initiation of a generic evaluation of the new analytical model by the organization responsible for methods and code review.

3. The predicted results of the transient are reviewed to ascertain that the values of pertinent system parameters are within expected ranges for the type and class of reactor under review. Further, the predicted results of the transient are reviewed to ensure that the consequences meet the acceptance criteria given in subsection II, below.

4. <u>Combined Operating License (COL) Action Items and Certification Requirements and Restrictions</u>. For a Design Certification (DC) application, the review will also address COL action items and requirements and restrictions (e.g., interface requirements and site parameters).

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

<u>Review Interfaces</u>

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. 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 applicant's technical submittal analysis are reviewed under DSRS Sections 4.2, 4.3, and 4.4.

4. Technical specifications are reviewed under DSRS Section 16.0.

5. Instrumentation and controls aspects of the sequence described in the applicant's technical submittal is reviewed to confirm that 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, interlocks with auxiliary or shared systems and compliance with Regulatory Guide 1.105 under DSRS Sections 7.0 through 7.2.

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 Chapter 19.

II. <u>ACCEPTANCE CRITERIA</u>

<u>Requirements</u>

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 17, Electric Power Systems. 5. GDC 26 Reactivity Control System Redundancy and Capability.

<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.

1. The basic objective in the review of the loss of normal feedwater transient is to confirm that the following criteria are met:

   A. The plant responds to the loss of feedwater transient in such a way that the criteria regarding fuel damage and system pressure are met.

   B. There is sufficient capacity for long term decay heat removal for the plant to reach a stabilized condition.

   C. The plant protection systems setpoints assumed in the transient analyses are selected with adequate allowance for measurement uncertainties as delineated in Regulatory Guide 1.105.

   D. The event evaluation takes into consideration single failures, operator errors, and performance of non-safety related systems that are consistent with regulatory guidelines set forth in RG 1.206.

2. Using the ANS standards as guidance, specific criteria have been developed to meet the relevant requirements of GDCs 10, 13, 15, 17, and 26 for events of moderate frequency and they are as follows:

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

   B. Fuel cladding integrity shall be maintained by ensuring that the minimum departure from nucleate boiling ratio (DNBR) remains above the 95/95 DNBR limit based on acceptable correlations (see applicant's technical submittal Section 4.4), as well as by satisfaction of any other specific acceptable fuel design limit (SAFDL) that may be applicable to the particular reactor design.

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

   D. To meet the requirements of GDCs 10 and 15, the positions of Regulatory Guide 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.

   E. 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 Regulatory Guide 1.53 and GDC 17.

   F. The guidance provided in SECY 77-439, SECY 94-084 and RG 1.206 with respect to the consideration of the performance of non-safety related systems during transients and accidents, as well as the consideration of single failures of active and passive systems (especially as they relate to the performance of check valves in passive systems) must be evaluated and verified.

3. The applicant's analysis of the loss of normal feedwater transient should be performed using an acceptable analytical model. 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 requests an evaluation by the appropriate organization which performs reactor systems reviews.

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

   A. The initial power level is taken as the licensed core thermal power for the number of loops initially assumed to be operating plus an allowance of 2% to account for power measurement uncertainties, unless a lower power level can be justified by the applicant. The number of loops operating at the initiation of the event should correspond to the operating condition which maximizes the consequences of the event.

   B. Conservative scram characteristics are assumed, i.e., the maximum time delay with the most reactive rod held out of the core, unless (a) a different conservatism factor can be justified through the uncertainty methodology and

evaluation, or (b) the uncertainty has otherwise been accounted for (see applicant's technical submittal Section 4.4).

C. The core burnup is selected to yield the most limiting combination of moderator temperature reactivity feedback, void reactivity feedback, Doppler reactivity feedback, power profile and radial power distribution.

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

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 Section 15.2.7 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. 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 DSRS Section 15.2.7 because this section evaluates the loss of normal feedwater flow transient. A part of the evaluation relates to the reactor coolant system being designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations including AOOs. Regulatory Guide 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 initiating events evaluated in this DSRS section involving a decrease in 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 instrument indications.

3. Compliance with GDC 15 requires that the reactor coolant system and associated auxiliary, control and protection systems shall 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 AOOs.

GDC 15 is applicable to DSRS Section 15.2.7 because this section evaluates the consequences of the events of a loss of normal feedwater flow transient that result in a decrease in heat removal by the secondary system with the potential for causing the reactor coolant system pressure to change in response to the increase 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 initiating events evaluated in this DSRS section involving a decrease in heat removal by the secondary system.

4. Compliance with GDC 17 requires (in part) that an onsite and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to ensure that SAFDLs and design conditions of the reactor coolant pressure boundary are not exceeded as a result of AOOs. GDC 17 is applicable to DSRS Section 15.2.7 because the loss of normal feedwater flow transient is an AOO.

   Meeting the requirements of GDC 17 provides assurance that SAFDLs and design conditions of the reactor coolant pressure boundary are not exceeded as a result of a loss of normal feedwater.

5. Compliance with GDC 26 requires that two independent reactivity control systems be provided capable of reliably controlling reactivity changes to ensure that acceptable fuel design limits are not exceeded.

   GDC 26 is applicable to DSRS Section 15.2.7 because this section evaluates the consequences of the events of a loss of normal feedwater flow that result in a decrease in heat removal by the secondary system with the potential for causing changes in reactivity within the core that could cause the thermal design criteria for the fuel cladding to be exceeded. DSRS 15.2.7 ensures that the thermal margin be sufficient to accommodate these conditions and ensures that the appropriate margins for malfunctions of reactivity controls such as stuck rods are accounted for.

   Meeting the requirements of GDC 26 provides assurance that SAFDLs are not exceeded by ensuring that there is 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 below are used for the design certification (DC) application review, the construction permit (CP), operating license (OL), and combined license (COL) applications. During the CP review the values of system parameters and setpoints used in the analysis will be preliminary in nature and subject to change. At the OL review stage, final values should be used in the analysis and the reviewer should compare these to the limiting safety system settings included in the proposed technical specifications.

The description of the loss of normal feedwater flow transient presented by the applicant in the applicant's technical submittal is reviewed by the organization responsible for reactor systems regarding the occurrences leading to the initiating event. The sequence of events from initiation until a stabilized condition is reached is reviewed to ascertain:

1. The extent to which normally operating plant instrumentation and controls are assumed to function.
2. The extent to which plant and reactor protection systems are required to function.
3. The extent to which credit is taken for the functioning of normally operating plant systems.
4. The extent to which operation of engineered safety systems is required.
5. The extent to which operator actions are required.
6. That appropriate margin for malfunctions, such as stuck rods, is accounted for.
7. The extent to which operation of auxiliary systems is required.
8. That instrumentation uncertainties of system and operating parameters are appropriately accounted for.

If the applicant's technical submittal states that the loss of feedwater transient is not as limiting as some other similar transient, the reviewer evaluates the justification presented by the applicant. If a quantitative analysis of the loss of feedwater transient is presented in the applicant's technical submittal, 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 loss of feedwater transient to an acceptable level. The reactor systems reviewer compares the predicted variation of system parameters with various trip and system initiation setpoints. The review of Chapter 7 of the applicant's technical submittal by the organization responsible for instrumentation and control systems confirms that their design is consistent with the requirements for safety systems actions for these events.

To the extent deemed necessary, the reactor systems reviewer evaluates the effect of single active failures of systems and components which may alter the course of the transient. For new applications, LOOP should not be considered a single failure; loss of feedwater should be analyzed with and without a LOOP in combination with a single active failure. This part of the review uses the procedures described in the DSRS sections for Chapters 4, 5, 6, 7, 8, and 9 of the applicant's technical submittal.

The mathematical models used by the applicant 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 reviews to determine if these models have been

previously reviewed and found acceptable by the staff. If not, a generic review of the model proposed by the applicant is initiated.

The values of system parameters and initial core and system conditions used as input to the model are reviewed by the organization responsible for reactor systems. Of particular importance are the reactivity feedbacks and control rod worths used in the applicant's analysis, and the variation of moderator temperature, void, and Doppler reactivity feedback with core life. The justification provided by the applicant to show that he has selected the core burnup that yields the minimum margins is evaluated.

The results of the analysis are reviewed, including the effects of the LOOP and the possibility of the event developing into a more serious event (e.g., a stuck open PORV on the pressurizer that could lead to a SBLOCA if not isolated), and compared with the acceptance criteria presented in subsection II of this DSRS section regarding maximum pressure in the reactor coolant and main steam systems. The parameters reviewed are:

1. variations with time during the transient of the neutron power,
2. heat fluxes (average and maximum),
3. reactor coolant system pressure,
4. minimum DNBR,
5. core coolant flow rates,
6. coolant conditions (inlet temperature, core average temperature, average exit and hot channel exit temperatures, and steam fractions),
7. steamline pressure,
8. containment pressure and temperatures,
9. pressure relief valve flow rate and quality,
10. pressurizer water volume, and
11. flow rate from the reactor coolant system to the containment system (if applicable).

The more important of these parameters for the loss of normal feedwater transient are compared with those predicted for other similar plants to see that they are within the range expected.

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 final safety analysis report (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 has reviewed the analyses of the loss of normal feedwater flow event and concludes that the analyses have adequately accounted for the operation of the plant and were performed using acceptable analytical models.

The staff further concludes that the analyses have demonstrated that the reactor protection and safety systems will continue to ensure that the SAFDLs and the RCPB pressure limits will not be exceeded as a result of the loss of normal feedwater flow.

The staff concludes that the plant design is acceptable with regard to transients resulting from loss of normal feedwater that are expected to occur with moderate frequency and that the predicted response meets the requirements of GDC 10, 13, 15, 17, and 26. This conclusion is based on the following:

The applicant has met the requirements of GDC 10 and 26 with respect to demonstrating that SAFDLs are not exceeded for this event and has met the requirements of GDC 15 with respect to demonstrating that the reactor coolant pressure limits have not been exceeded by this event.

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. The parameters used as input to the analytical model were reviewed and found to be suitably conservative and in accordance with the recommendation of Regulatory Guide 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 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.

Thus, the applicant has met the requirements of GDCs 17 and 26 with respect to demonstrating that SAFDLs are not exceeded for this event.

The applicant has met the positions of Regulatory Guide 1.53, SECY 77-439, SECY 94-084 and RG 1.206 as related to the single-failure criterion and Regulatory Guide 1.105 as related to instrument actuations of systems and components important to safety.

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

2. Regulatory Guide 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 50, "Domestic Licensing of Production and Utilization Facilities."

5. 10 CFR 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants."

6. 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).

7. ANSI/ANS 52.1-1983, "Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants," (replaced ANS Trial Use Standard N212-1974; reaffirmed 1988; withdrawn 1998).

8. General Design Criterion 10, "Reactor Design."

9. General Design Criterion 13, “Instrumentation and Control.”

10. General Design Criterion 15, "Reactor Coolant System Design."

11. General Design Criterion 17, "Electric Power Systems."

12. General Design Criterion 26, "Reactivity Control System Redundancy and Capability."

13. Regulatory Guide 1.53, "Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems."

14. Regulatory Guide 1.105, "Instrument Spans and Setpoints."

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

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

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

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