U.S. NUCLEAR REGULATORY COMMISSION

DESIGN-SPECIFIC REVIEW STANDARD for NuScale SMR DESIGN

15.1.1 - 15.1.4 DECREASE IN FEEDWATER TEMPERATURE, INCREASE IN FEEDWATER FLOW, INCREASE IN STEAM FLOW, AND INADVERTENT OPENING OF THE TURBINE BYPASS SYSTEM OR INADVERTENT OPERTION OF THE DECAY HEAT REMOVAL SYSTEM

REVIEW RESPONSIBILITIES

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

Secondary - None

I. AREAS OF REVIEW

Several events that are expected to occur with moderate frequency, and which involve an unplanned increase in heat removal by the secondary system, are covered by this design-specific review standard (DSRS) section. 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 could lead to a power level increase and a decrease in shutdown margin. The power level increase could lead to a reactor trip. Any unplanned power level increase could result in fuel damage or excessive reactor system pressure.

Each of the initiating events covered by this DSRS section should be discussed in individual sections of the safety analysis report (SAR) or the design control document (DCD), as specified in Regulatory Guide (RG) 1.70 and RG 1.206, "Combined License Applications for Nuclear Power Plants (Light-Water Reactor (LWR) Edition)."

The specific areas of review for the NuScale small modular reactor (SMR) are as follows:

1. feedwater system malfunctions that result in a decrease in feedwater temperature

2. feedwater system malfunctions that result in an increase in feedwater flow

3. steam pressure regulator malfunctions or failures that result in increased steam flow

4. inadvertent opening of the turbine bypass system

5. inadvertent actuation of the decay heat removal system (DHRS)

These feedwater system malfunctions and steam regulator malfunctions could result in increased cooling of the primary system water, which could result in an addition of positive reactivity. The inadvertent actuation of the DHRS could be caused by operator action or a false actuation signal that opens the valves that normally isolate the condenser from the reactor coolant system (RCS). This moderate-frequency event causes a cooldown of the RCS and results in the addition of positive reactivity in the presence of negative moderator feedback. The topics covered in the primary review include: postulated initial core and reactor conditions that are pertinent to feedwater system malfunctions, pressure regulator or pressure relief valve

ADAMS Accession No. ML15355A303

malfunctions, methods of thermal and hydraulic analysis, postulated sequence of events including time delays before 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 pressure, and reactivity.

The staff reviews the sequence of events described in the technical submittal for these transients. 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 condition. The analytical methods are reviewed to find out 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.

6. 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 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 DSRS 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 Standard Review Plan (SRP) Sections 4.2and 4.3 and DSRS Section 4.4.

6. Risk classifications are reviewed under SRP Section 19.3.

II. ACCEPTANCE CRITERIA

Requirements

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

1. 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 10, "Reactor Design"

2. GDC 13, "Instrumentation and Control"

3. GDC 15, "Reactor Coolant System Design"

4. GDC 20, "Protection System Functions"

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

6. Title 10 of the *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

DSRS Acceptance Criteria

Specific DSRS acceptance criteria acceptable to meet the relevant requirements of the U.S. Nuclear Regulatory Commission's (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 (for 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 and main steam systems 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. The guidance provided in RG 1.105, "Instrument Spans and Setpoints," can be used to analyze the effect of instrument spans and setpoints on the plant response to the type of transient addressed in this DSRS section, to meet the requirements of GDC 10, 13, 15, 20, and 26.

5. The most limiting plant system's 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, "Application of the Single-Failure Criterion to Safety Systems."

The applicant's analysis of transients caused by excessive heat removal should be performed using an acceptable analytical model, the NRC-approved methods and the computer codes. If the applicant proposes to use analytical methods that have not been approved, these methods are evaluated by the staff for acceptability. For new generic methods, the reviewer performs an evaluation based on SRP section 15.0.2, "Review of 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 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 that 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. Mitigating systems should be assumed to be actuated in the analyses at setpoints with allowance for instrument inaccuracy in accordance the guidance given in RG 1.105.

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 assure 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 secondary system, 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 by the secondary system.

2. GDC 13 requires the provision of instrumentation to monitor variables and systems over their anticipated ranges of normal operation, and of appropriate controls to maintain listed variables and systems within prescribed operating ranges.

   GDC 13 applies to this section because the reviewer evaluates the sequence of events, including automatic actuations of protection systems, and manual actions, and determines whether the sequence of events is justified, based upon the expected values of the relevant monitored parameters and instrument indications.

3. Compliance with GDC 15 requires that the reactor coolant system and associated auxiliary, control, and protection systems be designed with sufficient margin to assure 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 these overcooling events cause the reactor coolant system pressure to change in response to the drop in reactor coolant temperature. Although most of these events cause the reactor coolant pressure to decrease, some cause reactor coolant pressure to increase, depending on the worst single failure assumed. Therefore, for these overcooling transients of DSRS Section 15.1.1-15.1.4, the reactor coolant pressure needs to be analyzed to assure that the pressure acceptance criterion is satisfied.

   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 by the secondary system or inadvertent operation of the DHRS.

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

   GDC 20 is applicable to this section because the reviewer evaluates the reactor protection system that automatically shuts down the reactor to terminate the events (anticipated operational occurrences) evaluated 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 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 two required reactivity control systems use control rods capable of reliably controlling reactivity changes to assure 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 these overcooling events analyzed in this section that may involve 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 15.1.1-15.1.4 examines these margins where applicable to assure 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 acceptance criteria, the staff should review the applicant's evaluation of how the proposed alternatives offer an acceptable method of complying with the relevant NRC requirements identified in Subsection II.

1. Selected Programs and Guidance—In accordance with the guidance in NUREG–0800, "Introduction – Part 2: Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: Light-Water Small Modular Reactor Edition" (NUREG-0800, Intro Part 2), as applied to this DSRS Section, the staff will review the information proposed by the applicant to evaluate whether it meets the acceptance criteria described in Subsection II of this DSRS. As noted in NUREG-0800, Intro Part 2, the NRC requirements that must be met by a structures, systems, and components (SSC) do not change under the small modular reactor (SMR) framework. Using the graded approach described in NUREG-0800, Intro Part 2, the NRC staff may determine that, for certain SSCs, the applicant's basis for compliance with other selected NRC requirements may help demonstrate satisfaction of the applicable acceptance criteria for that SSC in lieu of detailed independent analyses. The design-basis capabilities of specific SSCs would be verified, where applicable, as part of completing the applicable ITAAC. The use of the selected programs to augment or replace traditional review procedures is shown in Figure 1 of NUREG-0800, Intro Part 2. Examples of such programs that may be relevant to the graded approach for these SSCs include:

   - 10 CFR Part 50, Appendix A, GDC, Overall Requirements, Criteria 1–5
   - 10 CFR Part 50, Appendix B, Quality Assurance (QA) Program
   - 10 CFR 50.49, Environmental Qualification (EQ) of Electrical Equipment Program
   - 10 CFR 50.55a, Code Design, Inservice Inspection, and Inservice Testing (ISI/IST) Programs

- 10 CFR 50.65, Maintenance Rule requirements
- Reliability Assurance Program (RAP)
- 10 CFR 50.36, “Technical Specifications”
- Availability Controls for SSCs Subject to Regulatory Treatment of Nonsafety Systems (RTNSS)
- Initial Test Program (ITP)
- Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC)

This list of examples is not intended to be all inclusive. It is the responsibility of the technical reviewers to determine whether the information in the application, including the degree to which the applicant seeks to rely on such selected programs and guidance, demonstrates that all acceptance criteria have been met to support the safety finding for a particular SSC.

2. In accordance with 10 CFR 52.47(a)(8), (21), and (22), and 10 CFR 52.79(a)(17), (20), and (37), for DC or COL applications submitted under 10 CFR Part 52, the applicant is required to (1) address the proposed technical resolution of unresolved safety issues and medium- and high-priority generic safety issues that are identified in the version of NUREG-0933, “Resolution of Generic Safety Issues,” current on the date up to 6 months before the docket date of the application and that 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 branch reviewing reactor systems 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 find out:

   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

   F. the 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 should 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 protections, 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 main steam lines are reviewed by the reactor systems 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.

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 reactivity and control rod worth used by the applicant in this analysis, and the variations of moderator temperature, void, and Doppler reactivity 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 main steam systems. 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. steam line pressure

   G. containment pressure

   H. pressure relief valve flow rate

I. flow rate from the reactor coolant system to the containment system (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 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 other COL action items; however, to ensure these COL action items are addressed during a COL application, they should be added to the DC technical submittal.

   For review of a COL application, the scope of the review is dependent on whether the COL applicant references a DC, an early site permit or other NRC approvals (e.g., manufacturing license, site suitability report or topical report).

IV. EVALUATION FINDINGS

The reviewer verifies that the applicant has provided sufficient information and that the staff's technical review and analysis, as augmented by the application of programmatic requirements in accordance with the 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.

A number of plant transients can result in an unplanned increase in heat removal by the secondary system or inadvertent operation of the DHRS. Those that might be expected to occur with moderate frequency can be caused by feedwater system or pressure regulator malfunctions, inadvertent opening of the turbine bypass system or inadvertent operation of the DHRS. All of these postulated transients have been reviewed. It was found that the most limiting transient in regard to core thermal margins and pressure within the reactor coolant and main steam systems was the _____ transient.

The staff concludes that the analysis of transients resulting in an unplanned increase in heat removal by the secondary system or inadvertent operation of the DHRS that are 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 we have determined that the applicant's analysis was performed using a mathematical model that had been previously 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, we have further determined that for applicants using the positions of RG 1.53 as related to the single failure criterion and RG 1.105 for instruments, that the methodology provided in those regulatory guides has 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 because 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 because the maximum pressure within the reactor coolant and main steam systems 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 SRP revision in effect 6 months before the docket date of the application. While the SRP provides generic guidance, the staff developed the SRP guidance based on the staff's experience in reviewing applications for construction permits and operating licenses for large light-water nuclear power reactors. The proposed SMR designs, however, differ significantly from large light-water nuclear power plant designs.

In view of the differences between the designs of SMRs and the designs of large light-water power reactors, the Commission issued Staff Requirements Memorandum (SRM)-COMGBJ-10-0004/COMGEA-10-0001, "Use of Risk Insights To Enhance Safety Focus of Small Modular Reactor Reviews," dated August 31, 2010 . In the SRM, the Commission directed the staff to develop risk-informed licensing review plans for each of the SMR design reviews, including plans for the associated preapplication activities. Accordingly, the staff has developed the content of the DSRS as an alternative method for evaluating a NuScale-specific application submitted pursuant to 10 CFR Part 52, and the staff has determined that each application may address the DSRS in lieu of addressing the SRP, with specified exceptions. These exceptions include particular review areas in which the DSRS directs reviewers to consult the SRP and others in which the SRP is used for the review. If an applicant chooses to address the DSRS, the application should identify and describe all differences between the design features (DC and COL applications only), analytical techniques, and procedural measures proposed in an application and the guidance of the applicable DSRS section (or SRP section, as specified in the DSRS), and discuss how the proposed alternative provides an acceptable method of complying with the regulations that underlie the DSRS acceptance criteria.

The staff has accepted the content of the DSRS as an alternative method for evaluating whether an application complies with NRC regulations for NuScale SMR applications, provided that the

application does not deviate significantly from the design and siting assumptions made by the NRC staff while preparing the DSRS. If the design or siting assumptions in a NuScale application deviate significantly from the design and siting assumptions the staff used in preparing the DSRS, the staff will use the more general guidance in the SRP, as specified in 10 CFR 52.17(a)(1)(xii), 10 CFR 52.47(a)(9), or 10 CFR 52.79(a)(41), depending on the type of application. Alternatively, the staff may supplement the DSRS section by adding appropriate criteria to address new design or siting assumptions.

VI. REFERENCES

1. U.S. Nuclear Regulatory Commission, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Regulatory Guide (RG) 1.70, November 1978, Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML011340072, ML011340108, and ML011340116.

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

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

4. *U.S. Code of Federal Regulations*, "Domestic Licensing of Production and Utilization Facilities," Part 50, Title 10, "Energy."

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. U.S. Nuclear Regulatory Commission, "Application of the Single Failure Criterion to Nuclear Power Plant Systems Protection," RG 1.53, November 2003, ADAMS Accession No. ML033220006.

11. U.S. Nuclear Regulatory Commission, "Instrument Spans and Setpoints," RG 1.105, November 1976, ADAMS Accession No. ML13064A112.

12. U.S. Nuclear Regulatory Commission, "Clarification of TMI Action Plan Requirements," NUREG-0737, November 1980, ADAMS Accession No. ML051400209.

13. U.S. Nuclear Regulatory Commission, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," NUREG-0800.

14. Staff Requirements Memorandum (SRM)-COMGBJ-10-0004/COMGEA-10-0001, "Use of Risk Insights To Enhance Safety Focus of Small Modular Reactor Reviews," dated August 31, 2010 (ML102510405)