U.S. NUCLEAR REGULATORY COMMISSION

DESIGN-SPECIFIC REVIEW STANDARD for NuScale SMR DESIGN

6.3 EMERGENCY CORE COOLING SYSTEM

REVIEW RESPONSIBILITIES

Primary - Organizations responsible for the review of the emergency core cooling system

Secondary - Organizations responsible for the review of other systems and technical areas related to the emergency core cooling system

The emergency core cooling system (ECCS) consists of two independent reactor vent valves (RVVs) and two independent reactor recirculation valves (RRVs). The ECCS provides a means of decay heat removal (DHR) in the event of a loss-of-coolant accident (LOCA) or a loss of the main feedwater flow in conjunction with the loss of both trains of the DHR system.

The ECCS removes heat and limits containment pressure by steam condensation on, and convective heat transfer to, the inside surface of the containment vessel. It allows heat conduction through the containment vessel walls and heat conduction and convection to the water in the reactor building pool. Long-term cooling is established via recirculation of reactor coolant to the reactor pressure vessel via the ECCS recirculation valves, which, when opened, provide a return flow of cooled water to the reactor.

The ECCS is initiated by opening the two RVVs exiting the top of the reactor pressure vessel (the pressurizer region) and the two RRVs entering the reactor pressure vessel in the downcomer region at a height above the core. Opening the valves allows a natural circulation path to be established. Water that is vaporized in the core leaves as steam through the RVVs, is condensed and collected in the containment vessel, and is then returned to the downcomer region inside the reactor vessel through the RRVs.

Following a LOCA or other condition resulting in an actuation of the ECCS, heat removal through the containment vessel rapidly reduces the containment pressure and temperature and maintains them at acceptably low levels for extended periods of time. Steam is condensed on the inside surface of the containment vessel, which is passively cooled by conduction and convection of heat to the reactor building pool water. The containment heat removal system is described in design-specific review standard (DSRS) Section 6.2.2.

I. <u>AREAS OF REVIEW</u>

The reviewer reviews the information presented in the applicant's technical submittal regarding the ECCS. The major elements of the review are as follows:

1. The design bases for the ECCS are reviewed to assure that they satisfy applicable regulations, including the applicable general design criteria (GDC) found in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix A, "General Design Criteria for Nuclear Power

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Plants”; the requirements of 10 CFR 50.46, “Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors,” regarding ECCS acceptance criteria; and Appendix K, “ECCS Evaluation Models,” to 10 CFR Part 50 regarding ECCS evaluation models.

2. The design basis for the ECCS is also reviewed for its capability to actuate automatically so as to assure timely and sufficient core coverage by cooling water in the event of a LOCA and the evaluation of the need for operator monitoring and manual actuation and control as a backup to automatic actuation.

3. For advanced passive reactors that rely on gravitational head to provide ECCS injection to the reactor coolant system (RCS), the RCS should be designed such that the available gravitational head is sufficient to provide adequate core cooling when depressurized.

4. For advanced reactors that rely on passive safety-related systems and equipment to automatically establish and maintain safe-shutdown conditions for the plant, these passive safety systems should be designed with sufficient capability to maintain safe-shutdown conditions for 72 hours, without operator actions and without nonsafety-related onsite or offsite power.

5. The design of the ECCS is reviewed to determine that it is capable of performing all of the functions required by the design bases.

6. The effects of accident-generated debris are reviewed, including an assessment for potential loss of the long-term cooling capability resulting from LOCA-generated and latent debris. Potential effects include debris screen blockage and debris fouling of nuclear fuel.

7. The preoperational and initial startup test programs for the ECCS are reviewed to determine if they are sufficient to confirm the performance capability of the ECCS. The need for special design features to permit the performance of adequate test programs should also be reviewed.

8. The proposed technical specifications (TS) are reviewed to assure that they are adequate in regard to limiting conditions of operation and periodic surveillance testing.

9. Inspections, Tests, Analyses, and Acceptance Criteria. For design certification (DC) and combined license (COL) reviews, the staff of the U.S. Nuclear Regulatory Commission (NRC) reviews the applicant’s proposed inspections, tests, analyses, and acceptance criteria (ITAAC) associated with the structures, systems, and components (SSCs) related to this DSRS section, in accordance with NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants” (SRP), Section 14.3, “Inspections, Tests, Analyses, and Acceptance Criteria.” The staff recognizes that the review of ITAAC cannot be completed until after the rest of the application has been reviewed against acceptance criteria contained in this DSRS section. Furthermore, the staff reviews the ITAAC to ensure that all SSCs in this area of review are identified and addressed as appropriate, in accordance with SRP Section 14.3 and DSRS Section 14.3.5.

10. COL Action Items and Certification Requirements and Restrictions. For a DC application, the review will also address COL action items and requirements and restrictions (e.g., interface requirements and site parameters).

    For a COL application referencing a DC, a COL applicant must address COL action items (referred to as COL 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. Evaluation of the ability of the ECCS to mitigate the consequences of a spectrum of LOCAs is performed under DSRS Section 15.6.5.

2. Evaluation of the capability of the applicant's mitigation of severe accidents is performed under SRP Section 19.0.

3. Review of the effects of pipe breaks outside containment on ECCS is performed under SRP Section 3.6.1. This review includes the evaluation of the effect of pipe whip, jet impingement forces, and environmental conditions.

4. Review of the acceptability of, and environmental qualification test program for, ECCS equipment is performed under DSRS Section 3.11. This review includes consideration of the postaccident environmental design and source term considerations described in Three Mile Island (TMI) action plan item II.B.2 of NUREG-0737, "Clarification of TMI Action Plan Requirements."

5. Review of the capability of those auxiliary systems essential for ECCS operation, spent fuel and reactor building pool cooling, clean-up system and ultimate heat sink (UHS) is performed under DSRS Section 9.1.3 and SRP Section 9.2.5. The evaluations include portions of the power conversion systems (e.g., steam supply lines, steam generators, feedwater systems) which interface with the RCS in such a way as to influence the course of a LOCA for a particular plant.

6. Review of the adequacy of ECCS-associated controls and instrumentation with regard to the features of automatic actuation, remote sensing and indication, and remote control is performed under DSRS Chapter 7. This review includes consideration of the instrumentation described in TMI action plan items II.D.3 (positive indication in the control room of flow in the discharge pipe of RCS relief and safety valves) and II.F.2 (instrumentation that provides an unambiguous, easy-to-interpret indication of inadequate core cooling) of NUREG-0737.

7. Review of the adequacy of containment isolation is performed under DSRS Section 6.2.4. Once the RVVs and RRVs open, containment becomes part of the ECCS. To provide recirculated water to maintain core cooling, the containment should be isolated to prevent inventory loss. The ability to isolate lines penetrating containment and thereby maintaining ECCS function is evaluated under DSRS Section 6.2.4.

8. Review of the adequacy of the power supply for the ECCS is performed under DSRS

Sections 8.1, 8.2, 8.3.1, and 8.3.2. In addition, a review of the plant's overall capabilities to withstand or cope with, and recover from, a station blackout (SBO) is performed under DSRS Section 8.4. The review of the adequacy of the power supply for the ECCS should coordinate with the review of the ECCS if the system is required to ensure adequate core cooling, as required by 10 CFR 50.63, "Loss of All Alternating Current Power." Regulatory Guide (RG) 1.155, "Station Blackout," provides guidance on coping during a SBO for a specified duration and transfer to an alternate alternating current (ac) source.

9. Review of the seismic and quality group classifications for the ECCS is performed under SRP Sections 3.2.1 and 3.2.2.

10. Review of the criteria used for postulating the effects of pipe breaks both inside and outside containment on ECCS is performed under SRP Section 3.6.2. This review includes criteria used for postulating the effects of pipe whip, jet impingement forces, and any related environmental conditions.

11. Review of the loading combinations (operational, LOCA, seismic, and thermal stratification loads) and the associated stress limits is performed under SRP Section 3.9.3.

12. Review of the adequacy of the Inservice Testing (IST) Program for valves is performed under SRP Section 3.9.6. This review is to assure that the ECCS piping and component configurations allow for full flow testing of safety-related valves and that provisions are made to allow for the use of advanced techniques to detect degradation and to monitor system performance.

13. Review of the structures housing the ECCS for the proper seismic classification is performed under DSRS Section 3.8.2 and SRP Section 3.9.3.

14. Review of the applicable inservice inspection (ISI) requirements is performed under DSRS Section 5.2.4 for Class 1 ECCS piping, if any, and components forming part of the reactor coolant pressure boundary (RCPB) and under DSRS Section 6.6 for Class 2 and 3 ECCS piping, if any, and components.

15. Review of any potential thermal shock effect of water recirculated into the primary coolant system from containment is performed, on a generic basis, under DSRS Sections 5.3.2 and 5.3.3.

16. Review of compliance with the requirements of item II.K.2.15 of NUREG-0737 is performed under DSRS Section 5.4.2.1. This review assures that once-through steam generator tubes are designed with sufficient margin to assure that, if the tubes are stressed under slug-flow conditions, mechanical integrity will be maintained.

17. Review of the proposed preoperational and initial startup test programs is performed under DSRS Section 14.2 to assure that they are consistent with the intent of RG 1.68, "Test Programs for Water-Cooled Nuclear Power Plants."

18. Review of compliance with the requirements of Task Action Plan items I.C.2 and I.C.6 of NUREG-0737 regarding procedures to assure that the system operability status is known is performed under SRP Sections 13.5.1.1 and 18.0.

19. Review of reliability and quality assurance (QA) is performed under SRP Sections 17.4 and 17.5.

20. Review of compliance with Task Action Plan item II.B.2 of NUREG-0737 and NUREG-0718 requirements regarding radiation and shielding design review is performed under SRP Section 12.1 and DSRS Sections 12.2 through 12.5 to assure adequate access to vital areas and protection of safety equipment.

21. Review of TS is performed under DSRS Section 16.0.

22. Review of the risk significance of SSCs is performed under SRP Section 19.0.

II. ACCEPTANCE CRITERIA

Requirements

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

1. GDC 2 as it relates to the seismic design of SSCs whose failure could cause an unacceptable reduction in the capability of the ECCS to perform its safety function

2. GDC 4 as it relates to dynamic effects associated with flow instabilities and loads (e.g., water hammer)

3. GDC 5 as it relates to SSCs important to safety not being shared among nuclear power units (NuScale reactor modules) unless it can be demonstrated that such sharing will not impair their ability to perform their safety function

4. GDC 17 as it relates to the design of the ECCS having sufficient capacity and capability to assure that specified acceptable fuel design limits and the design conditions of the RCPB are not exceeded during anticipated operational occurrences (AOOs) and that the core is cooled in the event of postulated accidents

5. GDC 27 as it relates to the ECCS design having the combined capability to assure that, under postulated accident conditions and with appropriate margin for stuck rods, the capability to cool the core is maintained

6. GDC 35, 36, and 37 as they relate to the ECCS being designed to provide an abundance of core cooling to transfer heat from the core at a rate so that fuel and clad damage will not interfere with continued effective core cooling, to permit appropriate periodic inspection of important components, and to permit appropriate periodic pressure and functional testing

7. 10 CFR 50.46, in regard to the ECCS being designed so that its cooling performance is in accordance with acceptable evaluation models, which identifies and accounts for uncertainties in the analysis method and inputs; alternatively, an ECCS evaluation model that may be developed in conformance with Appendix K to 10 CFR Part 50

8. TMI Action Plan item II.K.3.18 of NUREG-0737, which is equivalent to 10 CFR 50.34(f)(1)(vii) for applicants subject to 10 CFR 50.34(f), with respect to eliminating the need for manual actuation of the RVV to assure adequate core cooling

9. TMI Action Plan item II.K.3.28 of NUREG-0737, which is equivalent to 10 CFR 50.34(f)(1)(x) for applicants subject to 10 CFR 50.34(f), with respect to the RVV-associated equipment and instrumentation being capable of performing their intended functions during and following an accident, while taking no credit for nonsafety-related equipment or instrumentation, and accounting for normal expected air (or nitrogen) leakage through valves

10. 10 CFR 20.1406 as it applies to the ECCS and its subsystems for both normal operations and recovery from accident conditions with respect to how the facility design and procedures will minimize, to the extent practicable, contamination of the facility and the environment, facilitate eventual decommissioning, and minimize, to the extent practicable, the generation of radioactive waste

11. TMI Action Plan item II.D.3 of NUREG-0737, which is equivalent to 10 CFR 50.34(f)(2)(xi) for applicants subject to 10 CFR 50.34(f), with respect to the requirement that RCS relief and safety valves be provided with a positive indication in the control room of flow in the discharge pipe

12. TMI Action Plan item II.F.2 of NUREG-0737, which is equivalent to 10 CFR 50.34(f)(2)(xviii) for applicants subject to 10 CFR 50.34(f), with respect to the requirement that instrumentation or controls provide an unambiguous, easy-to-interpret indication of inadequate core cooling.

13. 10 CFR 52.47(b)(1), which requires that a DC application contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the DC is built and will operate in accordance with the DC, the provisions of the Atomic Energy Act (AEA) and the NRC's regulations

14. 10 CFR 52.80(a), which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the COL, the provisions of the AEA, and the NRC's regulations

15. 10 CFR 50.46(b)(5), as it relates to requirements for long-term cooling

<u>DSRS Acceptance Criteria</u>

Specific DSRS acceptance criteria that 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. In regard to the ECCS acceptance criteria of 10 CFR 50.46(b)(1), the five major performance criteria deal with:

   A. peak cladding temperature
   B. maximum calculated cladding oxidation
   C. maximum hydrogen generation
   D. coolable core geometry
   E. long-term cooling

   Guidance, procedures, and methods that are acceptable for meeting the requirements for a realistic or best-estimate evaluation model for ECCS performance can be found in RG 1.157, which is applicable only if NuScale or the COL chooses to use the option of using a realistic or best-estimate evaluation mode. This method should identify and account for uncertainties in the analysis method and inputs such that there is a high level of probability that the acceptance criteria are not exceeded (addresses NRC Generic Issue (GI) C-4). Alternatively, Appendix K to 10 CFR Part 50 contains guidance for conservative ECCS evaluation models. These areas are reviewed as a part of the effort associated with the LOCA analysis (DSRS Section 15.6.5). However, the impact of various postulated single failures on the operability of the ECCS, ECCS response times, break locations (including ECCS break locations), and break sizes affecting ECCS capabilities are evaluated under this DSRS section.

2. The NuScale ECCS must meet the requirements of GDC 35. The system must have alternative sources of Class 1E direct current (dc) electric power, as required by GDC 17, and must be able to withstand a single failure. The ECCS should retain its capability to cool the core in the event of a failure of any single active component (e.g., valves) during the short term immediately following an accident, or a single active or passive failure during the long-term recirculation cooling phase following an accident.

   A passive failure in a fluid system is a breach in the fluid pressure boundary or mechanical failure that adversely affects a flowpath. As stated in SECY-77-439 and SECY-94-084, "…the probability of most types of passive failures in fluid systems is sufficiently small that they need not be assumed in addition to the initiating failure in the application of single failure criterion to assure safety of a nuclear plant." Therefore, the approved position (SRM associated with SECY-94-084) is that fluid systems of passive advanced light-water reactor designs need not assume passive component failures in addition to the initiating failure in the application of a single-failure criterion to assure the safety of the plant. In addition, the staff considers, on a long-term basis, passive component failures in fluid as potential accident initiators, in addition to initiating events. Check valves in the passive safety systems (except those for which proper function can be demonstrated and documented) are considered components subject to single-failure consideration.

3. The ECCS must be designed to permit periodic ISI of important components, such as the RVVs, RRVs, reactor building pool, reactor pressure vessel, and containment vessel, in accordance with the requirements of GDC 36. The ECCS must be designed to permit testing of the operability of the system throughout the life of the plant, including the full operational sequence that brings the system into operation, as required by GDC 37.

4. The combined reactivity control system capability must meet the requirements of GDC 27 and should conform to the recommendation of RG 1.47. The primary mode of actuation for the ECCS should be automatic, and actuation should be initiated by signals of suitable diversity and redundancy. Provisions should also be made for manual actuation, monitoring, and control of the ECCS from the reactor control room.

5. Design features and operating procedures, designed to prevent damaging water hammer due to such mechanisms as voided discharge lines and water entrainment in steam lines, shall be provided to meet the requirements of GDC 4.

6. The design of those portions of the system that are not important to safety but the failure of which could have an adverse effect on the ECCS, must be in accordance with GDC 2, and acceptance is based on meeting Position C2 of RG 1.29. Also see SECY-94-084 and SECY-95-132 for policy and technical issues associated with the regulatory treatment of nonsafety systems in passive plant designs.

7. Interfaces between the ECCS and other cooling water systems should be such that operation of one does not interfere with, and provides proper support (where required) for, the other. In relation to these cooling water systems and other shared systems (e.g., containment heat removal system), the ECCS must conform to GDC 5.

8. To satisfy the requirement of 10 CFR 50.46(b)(5) regarding the long-term ECCS, the reactor building pool should be designed to provide a reliable, long-term DHR mechanism in conjunction with the ECCS. RG 1.82, Revision 4, provides guidance for post-LOCA long-term cooling.

9. The requirements and guidance for ECCS outage times and reports on ECCS unavailability, contained in Task Action Plan item II.K.3.17 and GI B-61, must also be satisfied. The GI is resolved by compliance with the Maintenance Rule (see below). Acceptable outage times for the ECCS are to be determined consistent with the assessment of risk significance of the ECCS based on the guidance of SRP Sections 16.1 and 19.

10. Programmatic requirements: Commission regulations and policy mandate “programs” applicable to SSCs including:

    - Maintenance Rule (SRP Sections 17.6 and 13.4 (Table 13.4, item 17))(discussed in RG 1.160)

    - TS (DSRS Section 16.0 and SRP Section 16.1)

    - Reliability Assurance Program (RAP) (SRP Section 17.4)

    - Availability controls (regulatory treatment for nonsafety systems (RTNSS)) (discussed in RG 1.206, Section C.IV.9)

    - Initial Plant Test Program (discussed in RG 1.68; DSRS Section 14.2; and SRP Section 13.4, Table 13.4, Item 19)

    - ITAAC (discussed in RG 1.215 and SRP Section 14.3)

In addition to the above criteria, the acceptability of the ECCS may be based on the degree of design similarity with previously approved plants.

Technical Rationale

The technical rationale for applying these acceptance criteria to the areas of review addressed by this DSRS section is discussed in the following paragraphs:

1. Compliance with GDC 2 requires, in part, that SSCs important to safety be designed to withstand the effects of natural phenomena without the loss of capability to perform their safety functions.

   GDC 2 is applicable, because the ECCS is relied upon to provide sufficient emergency core cooling flow to protect the integrity of the reactor core during postulated accidents, including a LOCA. RG 1.29 provides guidance for determining which SSCs should be designed to withstand the safe-shutdown earthquake (SSE). Position C.2 of RG 1.29 recommends that SSCs the continued function of which is not required but the failure of which could reduce the functioning of the ECCS to an unacceptable safety level should be designed and constructed to withstand the SSE.

   Meeting the requirement of GDC 2, and conforming to the positions of RG 1.29 provide assurance that plant safety is enhanced by ensuring the integrity of Seismic Category I portions of the ECCS and thus the capability to provide core cooling following a seismic event.

2. Compliance with GDC 4 requires, in part, that SSCs important to safety be designed to accommodate the effects of, and be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accident conditions. These conditions include consideration of the dynamic effects of flow instabilities and the loadings caused by water hammer events.

   GDC 4 is applicable, because the ECCS provides emergency core cooling in the event that normal cooling methods are not available or are insufficient.

   Meeting the requirements of GDC 4 provides assurance that dynamic effects of events such as flow instabilities and water hammer will not adversely affect the fundamental integrity and capability of the ECCSs to provide core cooling in the event of accidents.

3. Compliance with GDC 5 prohibits the sharing of SSCs among nuclear power units, unless it can be shown that such sharing will not significantly impair the ability of the SSCs to perform their safety functions, including, in the event of an accident in one module, an orderly shutdown and cooldown of the remaining modules.

   GDC 5 is applicable, because the ECCS provides an important safety function in its ability to provide emergency core cooling and shutdown capability following postulated accidents. The ECCS must be designed such that the ability to perform this and other designated safety-related functions are not compromised for each module, regardless of equipment failures or other events that may occur in another module.

Meeting the requirements of GDC 5 provides assurance that unacceptable effects of equipment failures or other events occurring in one module of a multimodule site will not propagate to the unaffected unit(s).

4. Compliance with GDC 17 requires that an onsite and an offsite electric power supply system be provided to permit functioning of SSCs important to safety.

   GDC 17 is applicable, as it relates to the ECCSs, because it requires that each power supply system have sufficient capacity and capability to ensure that the core can be cooled in the event of an accident and that the fuel design limits and the design conditions of the RCPB are not exceeded during AOOs. The ECCS may be dependent upon the availability of electrical power supplied from the Class IE emergency dc electrical busses. The power supplies for the ECCS should maintain voltages at electrical equipment within the design limits. With voltages below design limits, electric equipment may not have sufficient capacity or capability to reliably perform the intended safety function during a design-basis event.

   Meeting the requirements of GDC 17 enhances plant safety by ensuring that the ECCS capacity and capabilities will be sufficient to ensure that the fuel design limits and RCPB integrity are maintained during AOOs and that the core is cooled during accidents.

5. Compliance with GDC 27 establishes requirements regarding the combined reactivity control system capability.

   GDC 27 is applicable, because reliable control of reactivity is necessary to ensure reactor shutdown so that adequate core cooling can be provided by the ECCS. Control rod insertion, including appropriate margins for stuck rods, provides the negative reactivity to reduce reactor power to residual levels and ensures sufficient cooling flow to the core.

   Meeting the requirements of GDC 27 for the chemical and volume control system (CVCS) augments the protection for the primary fission product barrier by providing a means to ensure that the core, under postulated accident conditions, can be safely shut down and will be maintained in a coolable geometry.

6. Compliance with GDC 35 requires that an ECCS be provided that is capable of transferring heat from the reactor core, following a LOCA, at a rate sufficient to ensure that the core remains in a coolable geometry and that the clad metal-water reaction is limited to negligible amounts.

   GDC 35 is applicable, because following a breach in the RCPB, reactor coolant is lost at a rate determined by several factors, including break size and RCS pressure. The ECCS is relied upon to recirculate adequate cooling water into the RCS during a LOCA and to circulate the water through the core to provide core cooling. The ECCS must recirculate cooling water at a rate sufficient to ensure that the calculated changes in core geometry will be such that the core remains amenable to cooling and that the calculated cladding oxidation and hydrogen generation meet the specified performance criteria.

   Meeting the requirements of GDC 35 ensures that the ECCS, assuming a single failure, can provide core cooling under accident conditions sufficient to maintain the core in a coolable geometry and to minimize the production of hydrogen due to the reaction of water with the fuel cladding.

7. Compliance with GDC 36 requires that the ECCS be designed to allow for periodic inspections of important components to ensure the integrity and capability of the system.

   GDC 36 is applicable, because the ECCS arrangements must be designed such that adequate clearances are available to conduct periodic inspections of important components. Conduct of periodic inspections is necessary to show that important components of the ECCS are being maintained within their design-basis specifications and that no significant deterioration is occurring in the systems. Meeting the requirements of GDC 36 enhances plant safety by ensuring that important ECCS components can be inspected and will be capable of operating as designed to cool the core under accident conditions.

8. Compliance with GDC 37 requires that the ECCS be designed to allow for comprehensive periodic pressure and functional testing. GDC 37 is applicable, because the ECCS is required to undergo periodic pressure testing to verify the structural and leak-tight integrity of important components. Periodic functional testing of the ECCS verifies that the systems will operate as designed, including the full operational sequence necessary to initiate ECCS operation. Periodic functional test programs, such as valve testing, are premised upon the establishment of a reference set of parameters (based upon design specifications) and a consistent test method to allow for the detection of significant system degradation.

   Meeting the requirements of GDC 37 enhances plant safety by ensuring that important ECCS components can be tested and will remain capable of operating as designed to provide core cooling under postulated accident conditions.

9. Compliance with 10 CFR 50.46 requires that the ECCS be designed so that, in response to a LOCA, the calculated cooling performance is in accordance with an acceptable evaluation model or, alternatively, a model in conformance with the required and acceptable features of Appendix K to 10 CFR Part 50.

   The regulations in 10 CFR 50.46 are applicable, because the primary function of the ECCS is to provide emergency core cooling in the event of a LOCA resulting from a break in the primary RCS. The primary ECCS safety functions are comprehensively modeled and evaluated for breaks up to and including the largest opening in the RCPB to show that the ECCS will limit the peak clad temperature to below 1,204 degrees Celsius (2,200 degrees Fahrenheit) and ensure that the core will remain in place and substantially intact with its essential heat transfer geometry preserved.

   Meeting the requirements of 10 CFR 50.46, enhances plant safety by ensuring that the ECCS is designed and evaluated in such a way that the calculated core cooling performance after a LOCA conforms to critical criteria necessary to show that the core geometry will remain amenable to cooling and that long-term DHR will be provided.

III. <u>REVIEW PROCEDURES</u>

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: 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 an 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 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 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 which 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 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 relationship of the system under review to other previously approved plants is established. Systems or design features claimed to be identical or equivalent to those of previously approved plants are confirmed to be identical or equivalent.

4. Piping diagrams are reviewed to evaluate the functional reliability of the system in the event of single failures. That is, by referring to piping and instrumentation diagrams, the existence of the redundancy required by GDC 35 is confirmed.

5. The significant design parameters (e.g., water storage volume, system flow rate, and pressure) are examined for each component to confirm that these parameters satisfy operating requirements and the recommendations of RG 1.82 and 1.157.

6. The piping and instrumentation diagrams are checked to see that essential ECCS components are designated seismic Category I and Safety Class II (the cooling water side of heat exchangers can be Safety Class III).

7. The ECCS design is reviewed to confirm that the system can function in postaccident environments, considering possible mechanical effects; missiles; and the pressure, temperature, moisture, radioactivity, and chemical conditions resulting from a LOCA. Protection against valve motor flooding should be confirmed by the reviewer. Regarding such effects as pressure and temperature, the reviewer should confirm that accident conditions are specified that provide the basis for proof tests for the EQ of ECCS components.

8. The criteria, supporting analyses, plant design provisions, and operator actions that will be taken are reviewed to ensure that there will not be unacceptably high concentrations of boric acid in the core region (resulting in precipitation of a solid phase) during the long-term cooling phase following a postulated LOCA requiring actuation of the CVCS system.

9. The ECCS design is reviewed to confirm that there are provisions for maintaining the long-term coolant recirculation and DHR systems (e.g., valve overhaul) in the post-LOCA environment (including consideration of radioactivity).

10. The ECCS design is reviewed to confirm that opening the RVVs and RRVs does not occur prematurely (not enough inventory to provide for recirculation) and adversely affect the accident.

11. The availability of an adequate source of water for the ECCS (e.g., from the reactor) is confirmed, and the source volume, location, and susceptibility to failure are evaluated.

12. The RVVs and RRVs are reviewed using the following additional procedures to verify compliance with the acceptance criteria:

    A. The RVVs and RRVs, including electrical power supplies, are reviewed to verify they have sufficient independence, redundancy, and capability to allow the RVV/RRV to function properly, assuming a single failure.

    B. The RVV/RRV design is reviewed to verify that actuation of the system can be completed automatically and that manual actuation is not required to assure adequate core cooling.

    C. Design features and system analysis to verify performance of the RVV/RRV under all accident conditions are reviewed. The reviewer should verify the RVV/RRV can satisfy performance requirements without taking credit for nonsafety-related equipment or instrumentation, while accounting for normal air (or nitrogen) leakage through the valves.

    D. The applicant's evaluation of the RVV/RRV with respect to reactor vessel integrity limits is reviewed. If integrity limits could be exceeded during rapid cooldown, the applicant should evaluate alternative depressurization methods, other than full actuation of the RVV/RRV system, such as early depressurization with one or two relief valves.

    E. The RVV/RRV design is reviewed to verify that the valves are provided with a positive indication in the control room of flow in the discharge pipe.

    F. The ECCS (RRV and RVV) is part of the RCPB and should be designed to withstand pressure consistent with RCPB design criteria.

13. The reviewer should identify those portions of nonsafety-related systems that could have an adverse effect on the ECCS and should ensure that design provisions are in place to prevent these situations.

14. The ECCS is reviewed to evaluate the adequacy of design features that have been provided to prevent damaging water (steam) hammer, due to such mechanisms as voided discharge lines, water entrainment in steam lines, and steam bubble collapse.

    Guidance for water hammer prevention and mitigation is found in NUREG-0927.

15. The reviewer confirms that no component or feature of the ECCS in one reactor module is shared with the ECCS in another reactor module, or that shared features clearly meet the requirements of GDC 5.

16. The reviewer confirms that, within an individual reactor facility, any components shared between the ECCS and other systems (e.g., coolant makeup systems, residual heat removal systems, containment cooling systems) satisfy engineered safeguard feature design requirements and that the ECCS function of the shared component is not

diminished by the sharing.

17. The complete sequence of ECCS operation from accident occurrence through long-term core cooling is examined to see that a minimum of manual action is required and, where manual action is used, a sufficient time (greater than 20 minutes) is available for the operator to respond.

18. The reviewer confirms that long-term cooling capacity is adequate in the event of failure of any single active or passive component of the ECCS. If an intermediate heat transport system, such as the UHS, is used to provide long-term cooling capability, the system must be designed and constructed to an appropriate group classification, must be seismic Category I, and must be capable of sustaining a single active or passive failure without loss of function per GDC 2 and the 10 CFR 50 Appendix A definition of single failure.

19. With respect to ECCS power requirements, instrumentation and controls, and valve controls, the reviewer:

    A. confirms that the power requirements of the ECCS, including the timing of electrical loads, are compatible with the design of onsite emergency power systems

    B. confirms that there are sufficient instrumentation and controls available to the reactor operator to provide adequate information in the control room to assist in assessing post-LOCA conditions, including the more significant parameters, such as coolant flow, coolant temperature, and containment pressure

    C. confirms that automatic actuation and remote-manual valve controls are capable of performing the functions required; that suitable interlocks are provided, which do not impair separation of power trains or inhibit the required valve motions; and that instrumentation and controls have sufficient redundancy to satisfy the single failure criterion

20. Analyses are provided by the applicant in Chapter 15 of the design control document (DCD) to assess the capability of the ECCS to meet functional requirements. These analyses are reviewed, as described in DSRS Section 15.6.5, to determine conformance of the ECCS to the acceptance criteria. However, the review of the following portions of the ECCS response in LOCAs is performed by the reviewer under this DSRS section:

    A. The lower limit of reactor vessel leak size or interfacing pipe break size for which ECCS operation is required is established (i.e., the maximum leak or break size for which normal reactor coolant makeup systems can maintain the reactor pressure and coolant level is determined). The capability of the ECCS to actuate and perform at this lower limit of leak or break size is confirmed.

    B. The reviewer confirms that the analyses take into account a variety of potential locations for postulated reactor vessel leaks or interfacing pipe breaks, including the RRV and RVV outlets.

    C. The reviewer confirms that the analyses take into account a variety of single active failures. The reviewer should keep in mind that different single active failures may be limiting, depending on the particular reactor vessel leak or interfacing pipe break location and leak or break size postulated.

D. The ECCS component response times (e.g., for valves, power supply) are reviewed to confirm that they are within the delay times used in the accident analyses.

E. The ECCS design adequacy for all modes of reactor operation (e.g., full power, low power, hot standby, cold shutdown) is confirmed. The proposed plant TS are reviewed to

i. Confirm the suitability of the limiting conditions of operation, including the proposed time limits and reactor operating restrictions for periods when ECCS equipment is inoperable due to repairs and maintenance and that the means of indicating that safety systems have been bypassed or are inoperable are in accordance with RG 1.47

ii. Confirm that the limiting conditions for operation ensure that the specified operating parameters (e.g., minimum poison concentrations in the RCS and reactor building pool) are within the bounds of the analyzed conditions

iii. Verify that the frequency and scope of periodic surveillance testing is adequate

21. The reviewer verifies that the ECCS subsystems are designed to allow for comprehensive periodic ISI, pressure, and functional testing as indicated below:

A. The reviewer confirms that the design provides the capability for periodically demonstrating that the system will operate properly when an accident signal is received. That is, it should be demonstrated by the applicant that valves operate on normal and emergency power and that water pressure and flow are as designed when the plant is operating (periodic system surveillance). The ECCS design should have provisions to permit appropriate periodic inspection of important components and pressure testing.

B. The reviewer verifies that the design of the ECCS piping (if any) incorporates provisions to allow for full-flow testing (maximum design flow) of open isolation and check valves. For those designs where it is not practicable to conduct the IST at design flow and pressure, full-flow testing at maximum design flow with analysis to extrapolate to design pressure is sufficient.

C. The reviewer verifies that the ECCS design incorporates provisions to allow for testing the ECCS valves under design-basis differential pressure.

D. When it is not practicable to achieve design-basis differential pressure during ECCS valve testing, a qualification test (under design-basis differential pressure) before installation and inservice valve tests conducted under the maximum practicable differential pressure is sufficient.

22. The reviewer determines any special test requirements and confirms that the proposed preoperational test program for the ECCS is in conformance with the intent of RG 1.68.

23. The reviewer evaluates the applicant's description of how the ECCS design and procedures for operation will minimize, to the extent practicable, contamination of the facility and the environment; facilitate eventual decommissioning; and minimize, to the extent practicable, the generation of radioactive waste, in accordance with the provisions of 10 CFR 20.1406.

24. The reviewer verifies that the applicant has administrative procedures in place that establish limitations on the ECCS cumulative outage times (GI B-61). The reviewer verifies that the applicant will prepare and submit annual reports on ECCS unavailability (Task Action Plan II.K.3.17) that also include information on outage dates, lengths, and causes; ECCS components involved; and any corrective action taken. The reviewer verifies that acceptable outage times have been determined, consistent with assessing the risk significance of the ECCS, based on the guidance of SRP Sections 16.1 and 19.

25. The reviewer verifies that the applicant has reviewed its ECCS design configurations to identify any unisolable piping connected to the RCS that could be subjected to temperature distributions that would result in unacceptable thermal stresses. This review should consider the potential for thermal stratification, thermal cycling, and thermal fatigue, given the ECCS piping configurations. The reviewer verifies that appropriate action has been taken, where such piping is identified, to ensure that the piping will not be subjected to unacceptable thermal stresses. The review, which focuses on ECCS configurations, reviewing the stress analysis, and ensuring the stresses are in compliance with the ASME code, is performed under SRP Section 3.9.3.

26. The reviewer determines whether the ECCS allows for drainage of condensate from condensing accident-generated steam to the reactor building pool/UHS.

27. The reviewer verifies that full actuation of the ECCS does not lead to depressurization and cooldown that would exceed the vessel integrity limits.

28. 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), are set forth in 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 DCD.

    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).

29. For review of both DC and COL applications, SRP Section 14.3 should be followed for the review of ITAAC. The ITAAC review cannot be completed until after the completion of this section.

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 staff's technical review 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.

(For completeness, this evaluation finding includes the review effort described in DSRS Section 15.6.5.)

The ECCS includes the piping, valves, heat exchangers, instrumentation, and controls used to transfer heat from the core following a LOCA. The scope of review of the ECCS included system description and schematics, equipment layout drawings, failure modes and effects analyses, and design specifications for essential components. The review has included the applicant's proposed design criteria and design bases for the ECCS and the manner in which the design conforms to these criteria and bases.

The staff concludes that the design of the ECCS is acceptable and meets the requirements of GDC 2, 4, 5, 17, 27, 35, 36, and 37; 10 CFR 50.34(f)(1) (vii), (viii), (ix), (x) and 10 CFR 50.34(f)(2)(xi), (xviii); and 10 CFR 50.46. This conclusion is based on the following:

1. The applicant has met the requirements of GDC 2 with regard to the seismic design of nonsafety systems or portions thereof that could have an adverse effect on the ECCS by meeting position C.2 of RG 1.29 or an acceptable alternative.

2. The applicant has met the requirements of GDC 4 as related to the dynamic effects associated with flow instabilities and loads (e.g., water hammer).

3. The applicant has met the requirements of GDC 5 with respect to sharing SSCs by demonstrating that such sharing does not significantly impair the ability of the ECCS to perform its safety function, including, in the event of an accident to one unit, an orderly shutdown and cooldown of the remaining reactor modules.

4. The applicant has met the requirements of GDC 17 with regard to providing sufficient capacity and capability to assure that (a) specified acceptable fuel design limits and design conditions of the RCPB are not exceeded as a result of AOOs, and (b) the core is cooled and vital functions are maintained in the event of postulated accidents.

5. The applicant has met the requirements of GDC 27 with regard to providing combined reactivity control system capability to assure that, under postulated accident conditions and with appropriate margin for stuck rods, the capability to cool the core is maintained and the applicant's design meets the guidelines of RG 1.47 or an acceptable alternative.

6. The applicant has met the requirements of GDC 35 to provide abundant cooling for the ECCS by providing redundant safety-grade systems that meet the recommendations of RGs 1.82 and 1.157 or acceptable alternative methods.

7. The applicant has met the requirements of GDC 36 with respect to the design of the ECCS to permit appropriate periodic inspection of important components of the system.

8. The applicant has met the requirements of GDC 37 with respect to designing the ECCS to permit testing of the operability of the system throughout the life of the plant, including the full operational sequence that brings the system into operation.

9. The applicant has provided an analysis of the proposed ECCS relative to the acceptance criteria of 10 CFR 50.46, and with regard to the evaluation, models the guidance of RGs 1.82 and 1.157 or, alternatively, Appendix K of 10 CFR Part 50. The applicant has demonstrated that its ECCS designs satisfy the criteria for peak cladding temperature, maximum calculated cladding oxidation, maximum hydrogen generation, coolable core geometry, and long-term cooling, in accordance with an acceptable evaluation model.

10. The applicant has met item II.K.3.18 of NUREG-0737, equivalent to 10 CFR 50.34(f)(1)(vii) for applicants subject to 10 CFR 50.34(f), with respect to eliminating the need for manual actuation of the automatic depressurization system (ADS) (RVV) to assure adequate core cooling.

11. The applicant has met item II.K.3.28 of NUREG-0737, equivalent to 10 CFR 50.34(f)(1)(x) for applicants subject to 10 CFR 50.34(f), with respect to the ADS (RVV)-associated equipment and instrumentation being capable of performing their intended functions during and following an accident, while taking no credit for nonsafety-related equipment or instrumentation, while accounting for normal expected air (or nitrogen) leakage through valves.

12. The applicant has met the provisions of 10 CFR 20.1406 for ECCS and its subsystems for both normal operations and procedures and mechanisms for postaccident recovery.

13. The applicant has met the requirements of Task Action Plan item II.D.3 of NUREG-0737, equivalent to 10 CFR 50.34(f)(2)(xi) for applicants subject to 10 CFR 50.34(f), with respect to the requirements that RCS relief and safety valves be provided, in the control room, with a positive indication of flow in the discharge pipe.

14. The applicant has met the requirements of Task Action Plan item II.F.2 of NUREG-0737, equivalent to 10 CFR 50.34(f)(2)(xviii) for applicants subject to 10 CFR 50.34(f), with respect to the requirement that instrumentation or controls provide an unambiguous, easy-to-interpret indication of inadequate core cooling.

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.

In addition, to the extent that the review is not discussed in other SER sections, the findings will summarize the staff's evaluation of the ITAAC, including design acceptance criteria, as applicable.

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. 10 CFR 20.1406, "Minimization of Contamination."

2. 10 CFR 50.34(f), "Additional TMI-Related Requirements."

3. 10 CFR 50.46, "Acceptance Criteria for ECCS for Light Water Nuclear Power Reactors."

4. 10 CFR Part 50, Appendix A, GDC 2, "Design Bases for Protection Against Natural Phenomena."

5. 10 CFR Part 50, Appendix A, GDC 4, "Environmental and Dynamic Effects Design Basis."

6. 10 CFR Part 50, Appendix A, GDC 5, "Sharing of Structures, Systems, and Components."

7. 10 CFR Part 50, Appendix A, GDC 17, "Electric Power Systems."

8. 10 CFR Part 50, Appendix A, GDC 27, “Combined Reactivity Control System Capability.”

9. 10 CFR Part 50, Appendix A, GDC 35, “Emergency Core Cooling.”

10. 10 CFR Part 50, Appendix A, GDC 36, “Inspection of Emergency Core Cooling System.”

11. 10 CFR Part 50, Appendix A, GDC 37, “Testing of Emergency Core Cooling System.”

12. 10 CFR Part 50, Appendix K, “ECCS Evaluation Models.”

13. 10 CFR 52.47, “Contents of Applications.”

14. 10 CFR 52.80, “Contents of Applications; Additional Technical Information.”

15. Regulatory Guide 1.29, “Seismic Design Classification.”

16. Regulatory Guide 1.47, “Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems.”

17. Regulatory Guide 1.52, “Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants.”

18. RG 1.68, “Initial Test Programs for Water-Cooled Nuclear Power Plants.”

19. RG 1.79, “Preoperational Testing of Emergency Core Cooling Systems for Pressurized-Water Reactors.”

20. RG 1.82, “Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident.”

21. RG 1.155, “Station Blackout.”

22. RG 1.157, “Best-Estimate Calculations of Emergency Core Cooling System Performance.”

23. RG 1.160, “Monitoring the Effectiveness of Maintenance at Nuclear Power Plants.”

24. RG 1.206, “Combined License Applications for Nuclear Power Plants (LWR Edition).”

25. RG 1.215, “Guidance for ITAAC Closure Under 10 CFR Part 52.”

26. SECY-90-016, “Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements,” dated January 12, 1990.

27. SECY-93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs,” dated April 2, 1993.

28. Staff Requirements Memorandum, “SECY-93-087 - Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs,” dated July 21, 1993.

29. NRC Letter to all Holders of Operating Licenses or Construction Permits for Pressurized Water Reactors (PWRs), "Loss of Decay Heat Removal (Generic Letter No. 88-17)," October 17, 1988.

30. NRC Letter to All Holders of Light Water Reactor Operating Licenses and Construction Permits, "Guidance on Developing Acceptable Inservice Testing Programs (Generic Letter 89-04)," April 3, 1989.

31. NRC Bulletin 79-24, "Frozen Lines," September 27, 1979.

32. NRC Bulletin 88-08, "Thermal Stresses in Piping Connected to Reactor Coolant Systems," June 22, 1988, and its Supplements 1 through 3.

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

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

35. NUREG-0927, Revision 1, "Evaluation of Water Hammer Occurrences in Nuclear Power Plants," March 1984.

36. American National Standard, "Single Failure Criteria for PWR Fluid Systems," ANSI N658 (ANS 51.7).

37. NUREG-0933, "A Prioritization of Generic Safety Issues," July 1991.

38. NUREG-1462, "Final Safety Evaluation Report Related to Certification of the System 80+ Design."

39. SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-safety Systems in Passive Plant Designs."

40. SECY-95-132, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems (RTNSS) in Passive Plant Designs (SECY-94-084)."

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

42. NP-ER-0000-1198, "NuScale Plant Design Overview," August 13, 2012, Rev. 0 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12216A392)

43. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants."

44. NRC Staff Requirements Memorandum SRM-COMGBJ-10-0004/COMGEA-10-0001, "Use of Risk Insights To Enhance the Safety Focus of Small Modular Reactor Reviews," August 31, 2010, ADAMS Accession No. ML102510405.