Draft for Comment

U.S. NUCLEAR REGULATORY COMMISSION

DESIGN-SPECIFIC REVIEW STANDARD FOR NuScale SMR DESIGN

5.4.7 DECAY HEAT REMOVAL (DHR) SYSTEM

RESPONSIBILITIES

Primary - Organization responsible for review of reactor thermal hydraulic systems in SMRs

Secondary - None

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

The NuScale small modular reactor (SMR) makes extensive use of passive systems to meet regulatory requirements. Routine residual heat removal (RHR) for the NuScale SMR is provided by the main condenser and feedwater system and, at a lower temperature and pressure, filling the containment with water. Containment is the steel vessel surrounding the reactor pressure vessel and is one of the reactor modules located in the reactor building pool. Reactor coolant system (RCS) cooldown is achieved by conduction and convection of heat between the reactor pressure vessel wall, the water inside containment and the reactor building pool. During normal plant operations, heat is removed from the reactor building pool through a closed loop cooling system and ultimately rejected into the atmosphere through a cooling tower or other external heat sink. This function is further described in Design-Specific Review Standard (DSRS 9.2.5, Ultimate Heat Sink (UHS).

The Decay Heat Removal System (DHR) system provides secondary-side cooling for non-loss-of-coolant (LOCA) design basis events when normal secondary-side cooling is unavailable. The DHR function is included as a safety-related passive system for use during transients, and to provide a similar function as the auxiliary feedwater system (AFW) at a typical, large pressurized water reactor (PWR) in the event of a loss of main feedwater. The NuScale SMR safety-related RHR function is provided by the passive DHR, an engineered safety feature (ESF) of the NuScale design.

For transient conditions like loss of feedwater events, one DHR train is actuated by opening a pair of valves which are both located before (upstream of) the decay heat removal heat exchanger (DHRHX). The DHRHXs are both attached to the exterior of the containment vessel and are submerged in the reactor building pool. Once the DHR system is actuated, steam inside the steam generator tubes is redirected to the DHRHXs through a line connected to the main steam line upstream of the main steam isolation valves (MSIVs). The steam is then condensed in the DHRHXs and then returned back to the steam generator feedwater header via natural circulation.

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The review of the RHR function must consider all conditions from shutdown at normal RCS power operating pressure and temperature to the cold depressurized condition. Detailed evaluation of the cooling capacity of the DHR heat exchanger must also be considered for all anticipated operating conditions. The reviewer of this DSRS section will ensure that the design of the DHR conforms with General Design Criteria (GDCs) 1, 2, 4, 5,14, 19, 34 and 54 in Appendix A to Title 10 of the *Code of Federal Regulations* (CFR), Part 50 or similar requirements in the principle design criteria incorporated into the plant licensing basis.

The Pool Support Systems maintains reactor building pool water quality within limits for chemistry, radioactivity, and clarity. The reactor pool cooling system maintains reactor building pool water temperature and transfers pool water to the containment via the containment evacuation system during shutdown.

The Chemical and Volume Control System (CVCS) is used to provide makeup and letdown water to maintain the required water inventory and quality in the RCS and heats the reactor coolant during reactor start up before nuclear heat addition. These functions are reviewed in DSRS Section 9.3.6.

The active RHR function using the main steam system, feedwater system, and filling containment with water is used to cool the core during shutdown operations. High RHR function availability and reliability during shutdown conditions are important to mitigating risk and maintaining defense in depth. DSRS Section 9.3.6 covers the review of the methods used to ensure high reliability of the RHR function under these conditions.

NOTE: The RHR function is performed by RCI, and RCI is reviewed under DSRS 9.3.6. Additional review guidance is included in DCRS 9.3.6.

The reviewer will also evaluate the requirements for leakage detection and control identified in NUREG-0737, Item III.D.1.1. The organization responsible for the instrumentation and control systems reviews the hardware and procedures to provide reasonable assurance the leakage requirements are met.

Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC). For design certification (DC) and combined license (COL) reviews, the staff reviews the applicant's proposed ITAAC associated with the structures, systems, and components (SSCs) related to this DSRS section in accordance with Standard Review Plan (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 this portion 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.

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 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

The NuScale passive RHR function is provided by the safety-related DHR system and filling containment with water. The principle interfaces with this DSRS are as follows.

1. As part of its primary responsibility for SRP Section 3.12, the organization responsible for the review of materials engineering issues related to flaw evaluation and welding reviews the design of the RHR systems for new light-water reactor designs to verify, to the extent practical, that any low-pressure portions of the RHR that interface with the RCS will withstand full RCS pressure. If designing the RHR with ultimate rupture strength capable of withstanding full RCS pressure is not possible, the reviewer verifies that appropriate compensating measures have been taken in accordance with the review provided in SRP Section 3.12.

2. With respect to the staff review for compliance with Branch Technical Position (BTP) 5-4, the organizations responsible for the review of the steam and feedwater system, RCS, and reactor thermal hydraulic systems divide the evaluation as follows:

3. The organization responsible for the review of reactor thermal hydraulic systems in pressurized water reactors (PWRs) reviews the approach used to meet the functional requirements of BTP 5-4 with respect to cooling down to the conditions permitting operation of the DHR system. Since an alternate approach to that normally used for cooldown may be specified (for example, filling containment with water as an alternate to the use of the DHR system), the reviewer identifies all components and systems used. The organization responsible for the RCS has primary responsibility for the review of the pertinent portions of the chemical and volume control system (CVCS) (DSRS Section 9.3.4). The organization responsible for the containment and reactor building pool has primary responsibility for the review of the pertinent portions of the normal shutdown cooling system. The organization responsible for the review of the steam and feedwater systems has primary responsibility for DSRS Sections 10.3. The organization responsible for reactor thermal-hydraulic systems in pressurized-water reactors reviews the pressurizer relief valves and emergency core cooling system (ECCS). As part of its primary responsibility for DSRS Section 6.3, the organization responsible for the reactor thermal hydraulic systems reviews the iPWR depressurization systems used for cooldown. In addition, the organization responsible for the reactor thermal hydraulic systems reviews the tests and supporting analysis concerning the mixing of borated water and cooldown under natural circulation as required in BTP 5-1.

   A. The organization responsible for the review of reactor thermal hydraulic systems reviews the design and operating characteristics of the DHR with respect to its shutdown and long-term cooling function. Where the DHR interfaces with other systems (e.g., Pool Support Systems, UHS), the responsible organization reviews the effect of these systems on the DHR. The responsible organization also reviews overpressure protection provided by the interface (e.g., steam generator (SG) tube) between the RCS and DHR.

3. As part of its primary responsibility for DSRS Section 9.2.5, the organization responsible for the review of the ultimate heat sink (UHS) will coordinate with the reviewers of this section in order to verify that interaction between the DHR and the UHS is capable of:

   A. Providing sufficient reactor building pool level and cooling to the decay heat removal heat exchanger (DHRHX) in the event of DHR actuation; and,

   B. Maintaining and providing sufficient cooling to every DHR that shares the UHS with other adjacent DHRs of other 11 reactor modules. It must be verified that the sharing of the UHS between multiple reactor modules will not significantly impair the ability of the UHS to perform its safety function in the event of an accident in one module (unit), and orderly shutdown and cooldown of the remaining units (GDC-5).

Additionally, the organization responsible for the review of reactor thermal hydraulic systems will coordinate evaluations of other reviewers that interface with the overall review of the DHR as follows.

1. The organization responsible for the review of containment integrity performs the following reviews:

   A. Evaluates the containment heat removal capability and the reactor building pool designs as part of its review responsibility for DSRS Section 6.2.2; and,

   B. Verifies that portions of the DHR penetrating the containment barrier are designed with acceptable isolation features to maintain containment integrity for all operating conditions, including accidents, as part of its primary responsibility for DSRS Section 6.2.4 Note:

2. Review of flood protection for the DHR function is performed under DSRS Section 3.4.1. Flooding of the containment vessel from the reactor building pool is addressed under DSRS Section 9.1.3.

3. The organizations responsible for the structural analysis reviews and review of seismic/geotechnical issues determine the acceptability of the design analysis, procedures, and criteria used to establish the ability of seismic Category I structures housing the system and supporting systems to withstand the effects of natural phenomena such as a safe-shutdown earthquake (SSE), the probable maximum flood, and tornado missiles as part of their primary responsibility for DSRS Sections 3.3.1, 3.3.2, 3.5.3, 3.7.1 through 3.7.3, 3.8.4, and 3.8.5 and SRP 3.7.4. The organization responsible for the review of the inspection, testing, evaluation, and repair of mechanical equipment and components also verifies that inservice inspection requirements are met for system components as part of its primary responsibility for DSRS Section 6.6.

4. Upon request, the organization responsible for the review of component integrity issues related to engineered safety features verifies the compatibility of the materials of construction with service conditions as part of its primary responsibility for DSRS Section

6.1.1.

5. The organization responsible for mechanical engineering performs the following reviews:

   A . Determines the acceptability of the seismic and quality group classifications for system components as part of its primary responsibility for SRP Sections 3.2.1 and 3.2.2;

   B. Reviews the effects of pipe breaks inside and outside of containment if any, such as pipe whip and jet impingement, as part of its primary responsibilities for SRP Section 3.6.2;

   C. Determines that the components, piping, and structures are designed and tested in accordance with applicable codes and standards as part of its primary responsibility for SRP Sections 3.9.1, 3.9.2 and 3.9.3; and,

   D. Reviews adequacy of the inservice testing program of valves as part of its primary responsibility for SRP Section 3.9.6. The reviewer responsible for the reactor thermal hydraulic systems in SMRs should coordinate with the organization responsible for the mechanical engineering reviews to ensure that the DHR configuration allows 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 monitor system performance. Review of the ECCS passive decay heat removal system to ensure that residual heat will be removed during accident conditions is coordinated and performed under DSRS Section 6.3.

6. The organization responsible for quality assurance performs the following reviews:

   A. Evaluates the pre-operational and startup test programs to confirm that they are in conformance with the intent of Regulatory Guide (RG) 1.68 as part of its primary responsibility for DSRS Section 14.2

   B. Has primary responsibility for Task Action Plan items I.C.2 and I.C.6 of NUREG-0737 regarding procedures to ensure that system operability status is known, as part of its review responsibility for SRP Section 13.5.1.1

   C. Evaluates quality assurance as part of its primary responsibility for SRP Chapter 17. Reviews cooling water systems that transfer decay heat from the DHR to atmosphere is performed under DSRS Section 9.2.2.

7. The organization responsible for the review of the mechanical effects of missiles on SSCs performs the following reviews:

   A. Evaluates flood protection as part of its primary responsibility for DSRS Section 3.4.1.

   B. Identifies the SSCs to be protected against externally generated missiles and

reviews the adequacy of protection against such missiles as part of its primary responsibility for DSRS Sections 3.5.1.4 and 3.5.2.

C. Reviews protection against internally generated missiles both inside and outside of containment as part of its primary responsibility for DSRS Sections 3.5.1.1 and 3.5.1.2. Review of steam and feedwater systems that support initiation of normal plant cooldown is performed under DSRS Sections 10.3 and 5.4.14.

8. The organization responsible for the review of proposed preoperational and initial startup test programs ensures that DHR function will remove core residual heat as part of its primary responsibility for DSRS Section 14.2.

9. The organization responsible for the review of the UHS associated with the balance of plant reviews the plant design for protection against postulated piping failures outside containment, as part of its primary responsibility for SRP Section 3.6.1.

10. The organization responsible for the review of environmental qualification of electrical equipment reviews the acceptability of, and environmental qualification test program for, DHR equipment exposed to a post-accident environment, including consideration of the post-accident environmental design and source term considerations described in NUREG-0737 Task Action Plan Item II.B.2 and NUREG-0718, as part of its review responsibility for DSRS Section 3.11.

11. The organization responsible for the review of fire protection performs a review of fire protection as part of its primary responsibility for DSRS Section 9.5.1.

12. The organization responsible for the electrical engineering and power systems reviews identifies the safety-related electrical loads and determines that power systems supplying motive or control power for the DHR meet acceptable criteria and will perform these intended functions during all plant operating and accident conditions, as part of its primary responsibility for DSRS Sections 8.1, 8.2, 8.3.1, and 8.3.2. In addition, this organization reviews the capability to withstand or cope with and recovers from a station blackout and coordinates with the review of the DHR if the system is required to ensure adequate core cooling and/or decay heat removal, as part of its review under DSRS Section 8.4. If the power supply (DC) to the DHR valves is from a RTNSS Battery system, the review will be performed under DSRS Section 8.

13. The organization responsible for the review of instrumentation and control systems (permissive and interlocks) reviews those systems for the DHR to determine that it will perform its design function as required and conform to all applicable acceptance criteria, as part of its primary responsibility for DSRS Sections 7.1 and 7.2. This organization also reviews the provisions to meet GDC 19 with respect to equipment outside of the control room for hot and cold shutdown.

14. The organization responsible for the review of health physics has primary responsibility for DSRS Sections 12.1 through 12.5, including conformance with NUREG-0737 Task Action Plan Item II.B.2 and NUREG-0718, which involve a

radiation and shielding design review and corrective actions to ensure adequate access to vital areas and protection of safety equipment.

15. The organization responsible for the review of applicable technical specifications evaluates the technical specifications as part of its primary responsibility for DSRS Section 16.0.

16. Review of the reliability assurance program (RAP) is coordinated and performed under SRP Section 17.4.

II. ACCEPTANCE CRITERIA

Requirements

Acceptance criteria are based on meeting the relevant requirements of the following Commission regulations. The specific areas of review for the safety-related functions of the DHR are as listed below.

1. 10 CFR Part 50, Appendix A, General Design Criteria (GDC) 1, Quality Standards and Records.

2. GDC 2, Design Bases for Protection Against Natural Phenomena.

3. GDC 4, Environmental and Dynamic Effects Design Bases.

4. GDC 5, Sharing of Structures, Systems, and Components.

5. GDC 14, Reactor Coolant Pressure Boundary.

6. GDC 19, Control Room.

7. GDC 34, Residual Heat Removal.

8. GDC 44, Cooling Water.

9. GDC 45, Inspection of cooling water system.

10. GDC 46, Testing of cooling water system.

11. GDC 54, Piping Systems Penetrating Containment.

12. GDC 57, as it relates to closed system isolation valves on piping systems penetrating primary reactor containment.

13. NUREG-0737 Task Action Plan Item III.D.1.1, correlating to 10 CFR 50.34(f)(2)(xxvi), for applicants subject to 10 CFR 50.34(f), as it relates to the provisions for a leakage detection and control program to minimize the leakage from those portions of the DHR system

outside of the containment that contain or may contain radioactive material following an accident.

14. 10 CFR 50.62, as it relates to the design provisions for automatic initiation of the auxiliary feedwater system (for NuScale, this would be the DHR) in an anticipated transient without scram (ATWS) event.

15. 10 CFR 50.63, as it relates to the design provisions for withstanding and recovering from a station blackout, including an acceptable degree of independence from the ac power system and the capability for removal of decay heat at an appropriate rate for an appropriate duration.

16. 10 CFR 52.47(b)(1), which requires that a DC application contain the proposed ITAACs 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 U.S. Nuclear Regulatory Commission's (NRC's) regulations.

17. 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.

18. 10 CFR 20.1406, which requires that facility 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.

DSRS Acceptance Criteria

Specific DSRS acceptance criteria acceptable to meet the relevant requirements of the NRC's regulations identified above are set forth below. The DSRS is not a substitute for the NRC's regulations, and compliance with it is not required. As an alternative, and as described in more detail below, an applicant may identify the differences between a DSRS section and the design features (DC and COL applications only), analytical techniques, and procedural measures proposed in an application and discuss how the proposed alternative provides an acceptable method of complying with the NRC regulations that underlie the DSRS acceptance criteria.

The specific areas of review for the safety-related functions of RCI are as listed below.

1. The NuScale RHR function should satisfy the functional, isolation, pressure relief, and test requirements specified in Branch Technical Position (BTP) 5-4 and GDC 1.

2. The safety-related RHR function should be capable of withstanding the effects of natural phenomena like earthquakes, tornadoes, hurricanes, and floods to meet the requirements of GDC 2.

3. To meet the requirements of GDC 4, Design features and operating procedures should be provided for the RHR function to prevent damaging water hammer caused by such mechanisms as voided lines.

4. To meet the requirements of GDC 5, SSCs performing a safety function must not be shared between units of a multi-unit site, unless it can be shown that such sharing will not significantly impair the ability of the SSCs to perform their safety functions.

5. Extremely low failure probability of the reactor coolant pressure boundary must be demonstrated to meet the requirements of GDC 14.

6. Controls should be available to the operator to provide for RHR from the control room and an alternate shutdown location outside the control room during normal and shutdown operations as required by GDC 19.

7. Operation of the risk-significant RHR function should be administratively controlled by procedure and coordinated with the operation of the ECCS passive decay heat removal system to provide for RHR from the reactor core at a rate such that specified fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as required by GDC 34.

8. Piping systems penetrating primary reactor containment must be provided with leak detection, isolation, and containment capabilities to meet the requirements of GDC 54.

9. Interfaces between the DHR system and other systems should be designed so that operation of one does not interfere with, and provides proper support (where required) for, the other. In relation to these and other shared systems (e.g., emergency core cooling and containment heat removal systems), the DHR system must conform to GDC 5 (as it relates to understanding maximum pool temperature determination for multiple unit affects (i.e., multiple shutdowns, multiple shutdowns plus one loss of coolant accident)).

10. The guidelines of RG 1.82 regarding water sources for long term cooling following a loss-of-coolant accident should be considered in the design.

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. The specific areas of review for the safety-related DHR functions are as listed below.

1. GDC 1 requires that SSCs important to safety be designed, fabricated, erected and tested to quality standards commensurate with the importance of the safety functions to be performed. The DHR is important to safety in that: 1) the DHR is relied upon to provide an auxiliary feedwater-like function in the event that the main feedwater system is not in service or unavailable; 2) through connections to the RCS (e.g., SG tube break or collapse), a DHR failure could adversely affect the integrity of the RCS or containment systems; and 3) portions of the DHR can contain radioactive material in the event of an accident. Meeting the requirements of GDC 1 (and the guidance of RG 1.26) ensures that the DHR will be designed, fabricated, erected and tested to generally accepted and recognized codes and standards that are sufficient to assure a quality system in keeping with the importance of the designated safety functions.

2. GDC 2 requires that SSCs important to safety be designed to withstand the effects of natural phenomena, such as earthquakes, without the loss of capability to perform their safety functions. The NuScale DHR function is relied upon to provide defense in depth to the ECCS passive decay heat removal system for the removal of residual heat from the reactor core to maintain the reactor in a safe-shutdown condition. RG 1.29 provides guidance for determining which systems should be designated Seismic Category I; Regulatory Position C.1 provides guidance for safety-related portions and Regulatory Position C.2 addresses nonsafety-related systems and components. Meeting the requirements of GDC 2 will enhance plant safety by ensuring that the DHR function will be available to cool the core during and following an external event.

3. GDC 4 requires 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, including such effects as pipe whip and jet impingement. The safety-related RHR function of the NuScale DHR is to transfer heat from the reactor to the environment during and after plant shutdown. To ensure the availability of the decay heat removal function, the DHR must be capable of performing heat transfer under the expected operational and postulated transient conditions for the plant. Heat transfer during postulated accident conditions for the plant is a function of the passive ECCS decay heat removal system, evaluated in DSRS Section 6.3. NuScale is crediting the DHR for Chapter 15 accidents that involve a loss of secondary side heat removal such as turbine trip and loss of feedwater.

   These conditions include consideration of the dynamic effects of flow instabilities and the loadings caused by steam or water hammer events. Compliance with GDC 4 enhances plant safety by providing assurance that the dynamic effects of events such as flow instabilities and steam or water hammer will not affect the capability of the NuScale DHR function to remove decay heat during operational and transient conditions.

4. 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 unit, and orderly shutdown and cooldown of the remaining units. The DHRs are relied upon to transfer decay heat from the reactor to the environment after a reactor shutdown. The DHR must be designed such that the ability to perform the safety-related functions is not compromised for each unit among the multiple reactor modules regardless of equipment failures or other events that

may occur in another unit. Meeting the requirements of GDC 5 enhances plant safety by providing assurance that the unacceptable effects of equipment failures or other events occurring in one module (unit) of a multiple reactor module site will not prevent an orderly shutdown and cooldown of the unaffected unit(s).

5. GDC 14 requires assurance that the reactor coolant pressure boundary (RCPB) will have an extremely low probability of abnormal leakage, of rapidly propagating failure and of gross rupture (e.g., SG tube rupture). Failure of the RCPB (e.g., SG tube) may be postulated where the mechanisms of general corrosion and/or stress corrosion cracking induced by impurities in the reactor coolant are present. The Chemical and Volume Control (CVC) system maintains acceptable purity levels in the reactor coolant through the removal of insoluble corrosion products and dissolved ionic material by filtration and ion exchange. In addition, the CVC maintains proper RCS chemistry by controlling total dissolved solids, pH, oxygen concentration, and halide concentrations within the acceptable ranges. Meeting the requirements of GDC 14 enhances facility safety by providing assurance that the probability of corrosion-induced failure of the RCPB (Steam Generator tubes) will be minimized, thereby maintaining the integrity of the RCPB.

6. GDC 19 requires that a control room be provided from which actions can be taken to operate the nuclear power unit during both normal operating and accident conditions, including the loss-of-coolant accident (LOCA). BTP 5-4 provides guidance for compliance with GDC 19 with regard to achieving cold shutdown from the control room using only safety-grade equipment. The NuScale DHR function may be used for safe shutdown and cooldown of the reactor during normal and transient conditions. Compliance with GDC 19 enhances plant safety by ensuring the availability of adequate instrumentation and controls in the control room to perform the required safety-related functions of the DHR function under all anticipated conditions.

7. GDC 34 requires the capability to transfer decay heat and other residual heat from the reactor such that fuel and pressure boundary design limits are not exceeded. In addition, the system must be designed with sufficient redundancy and isolation capability to ensure that the safety function can be accomplished assuming a single failure of an active component with or without a coincident loss of offsite power. The DHR system transfers the fission product decay and other residual heat from the reactor core.

   Removal of decay and residual heat is necessary to prevent core damage under both normal and accident shutdown conditions. BTP 5-4 provides an acceptable approach to ensure compliance with GDC 34 with regard to accomplishing the DHR system safety functions assuming a single failure. Compliance with GDC 34 enhances plant safety by providing assurance that decay and RHR will be accomplished and the RCS pressure boundary and fuel cladding integrity will be maintained, thereby minimizing the potential for the release of fission products to the environment.

8. GDC 44 requires a system to transfer heat from structures, systems, and components important to safety to an ultimate heat sink. The system safety function must be to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions. Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to

assure that the system safety function can be accomplished, assuming a single failure of an active component with or without a coincident loss of offsite power. The DHR, by itself transfers, heat from structures, systems, and components important to safety to the ultimate heat sink, i.e. the reactor building pool, via the DHRHXs. The NuScale design incorporates two 100% capacity trains of DHR per reactor, allowing for suitable redundancy to assure the safety function can be accomplished with either offsite or onsite power available, concurrent with a single failure. Compliance with GDC 44, transferring of heat from SSCs important to safety to the UHS and having suitable redundancy in the DHR components and features provides a high level of assurance that SSCs important to safety do not overheat and fail in any normal operating or accident condition.

9. GDC 45 requires a cooling water system (here the DHR) that is designed for appropriate periodic inspection of important components like heat exchangers and piping to assure system integrity and capability. By periodic monitoring to detect signs of system degradation or incipient failure, GDC 45 provides assurance that the DHR will function reliably to provide decay heat removal and essential cooling to the reactor core.

10. GDC 46 requires a cooling water system (here the DHR) that is designed for appropriate periodic pressure and functional testing of the structural and leak-tight integrity of its components, the operability and the performance of system active components, and the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation for reactor shutdown and for loss-of-coolant accidents, including operation of portions of the protection system and transfer between normal and emergency power sources. By designing the DHR for testing to detect degradation in performance or in the system pressure boundary, GDC 46 assures that the DHR will function reliably to provide decay heat removal and essential cooling to the reactor core.

11. GDC 54 requires that piping systems that penetrate primary reactor containment be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems must be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits. Piping in the DHR passes through the containment boundary and is provided with isolation valves and integrity verification capabilities. Compliance with GDC 54's containment isolation and leak detection requirements provides a high level of assurance that the containment will perform its safety function in the event of a postulated accident and will maintain the capability to prevent a significant uncontrolled release of radioactivity.

12. GDC 57 requires that a closed system shall have at least one containment isolation valve and that valve shall be located as close to the containment as practical.

13. 10 CFR 50.62 requires that pressurized water reactors have equipment diverse from the reactor protection system to initiate the DHR under conditions indicative of an ATWS. The DHR is required to assure adequate removal of heat from the reactor coolant system during an ATWS.

14. 10 CFR 50.63 requires that all light-water-cooled nuclear power plants be able to withstand and recover from a station blackout. RG 1.155 provides guidance for compliance with 10 CFR 50.63. As many safety systems necessary to remove decay heat from the reactor are dependent on AC power, station blackout consequences can be severe. In a station blackout at facilities where no alternate AC power source is provided, the capability to cool the reactor core depends on the availability of systems not reliant upon AC power or on the ability to restore AC power in a timely manner. The DHR is required for removal of decay heat in the event of a station blackout and must have sufficient capability and capacity to perform the heat removal function for an appropriate duration. To ensure such capability, motive power for motors, valves, controls and instrumentation for at least one DHR train of adequate capacity for station blackout is verified to be provided that is independent of the normal and emergency AC power systems. Compliance with 10 CFR 50.63 and the positions of RG 1.155 on capability to withstand or cope with a station blackout provides protection against unacceptable offsite radiological consequences should both offsite and onsite emergency AC power systems fail concurrently.

15. 10 CFR 20.1406(a), requires that a DC or COL applicant describe how facility 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. 10 CFR 20.1406(a) applies to this DSRS section because the possibility of leakage of radioactive water in the Steam Generator exists.

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.

For design control document (DCD) and COL reviews, the reviewer uses the procedures to verify that the final design appropriately implements the initial design criteria and bases as set forth in the final safety analysis report (FSAR) technical submittal. The COL review also covers the proposed technical specifications to ensure that they are adequate with regard to limiting conditions of operation and periodic surveillance testing.

The review includes all of the systems used to transfer residual heat from the reactor over the entire range of potential reactor coolant temperatures and pressures.

1. Selected Programs and Guidance - In accordance with the guidance in NUREG-0800, "Introduction - Part 2: Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: Integral Pressurized Water Reactor Edition" (NUREG-0800 Intro Part 2) as applied to this DSRS Section, the staff will review the information proposed by the applicant to evaluate whether it meets the acceptance criteria described in Subsection II of this DSRS. As noted in NUREG-0800 Intro Part 2, the NRC requirements that must be met by an SSC do not change under the SMR framework. Using the graded approach

described in NUREG-0800 Intro Part 2, the NRC staff may determine that, for certain structures, systems, and components (SSCs), the applicant's basis for compliance with other selected NRC requirements may help demonstrate satisfaction of the applicable acceptance criteria for that SSC in lieu of detailed independent analyses. The design-basis capabilities of specific SSCs would be verified where applicable as part of completion of the applicable ITAAC. The use of the selected programs to augment or replace traditional review procedures is described in Figure 1 of NUREG-0800, Introduction - Part 2. Examples of such programs that may be relevant to the graded approach for these SSCs include:

- 10 CFR Part 50, Appendix A, General Design Criteria (GDC), Overall Requirements, Criteria 1 through 5
- 10 CFR Part 50, Appendix B, Quality Assurance (QA) Program
- 10 CFR 50.49, Environmental Qualification of Electrical Equipment (EQ) Program
- 10 CFR 50.55a, Code Design, Inservice Inspection and Inservice Testing (ISI/IST) Programs
- 10 CFR 50.65, Maintenance Rule requirements
- Reliability Assurance Program (RAP)
- 10 CFR 50.36, Technical Specifications
- Availability Controls for SSCs Subject to Regulatory Treatment of Non-Safety Systems (RTNSS)
- Initial Test Program (ITP)
- Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC)

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

2. In accordance with 10 CFR 52.47(a)(8),(21), and (22), and 10 CFR 52.79(a)(17), (20) and (37), for design certification or combined license applications submitted under Part 52, the applicant is required to (1) address the proposed technical resolution of unresolved safety issues and medium- and high-priority generic safety issues which are identified in the version of NUREG-0933 current on the date up to 6 months before the docket date of the application and which are technically relevant to the design; (2) demonstrate how the operating experience insights have been incorporated into the plant design; and, (3) provide information necessary to demonstrate compliance with any technically relevant portions of the Three Mile Island requirements set forth in 10 CFR 50.34(f), except paragraphs (f)(1)(xii), (f)(2)(ix), and (f)(3)(v) for a DC application, and except paragraphs (f)(1)(xii), (f)(2)(ix), (f)(2)(xxv), and (f)(3)(v) for a COL application. These cross-cutting review areas should be addressed by the reviewer for each technical subsection and relevant conclusions documented in the corresponding safety evaluation report (SER) section.

3. Using the description given in the applicant's DCD technical submittal, including

component lists and performance specifications, the reviewer determines that the system DHR piping and instrumentation provide reasonable assurance that the system(s) will operate as intended, with or without available offsite power and given any single active component failure. To do this, the reviewer evaluates the piping and instrumentation diagrams (P&IDs) or equivalent figures, system description and schematics to confirm that piping arrangements permit the achievement of the required flow paths and that sufficient process sensors are available to measure and transmit required information. The reviewer uses a failure modes and effects analysis (or similar system safety analysis) provided in the DCD to determine conformance to the single failure criterion.

4. Where possible, comparisons should be made with actual performance data from similar systems in operating plants. The reviewer can use previously reviewed RHR designs as a guide, but must verify that any differences are justified.

5. From the system description and P&IDs, the reviewer determines that the isolation requirements of BTP 5-4 are satisfied.

6. The staff reviews the DHR function to evaluate the adequacy of design features and procedures that have been provided to prevent damaging steam or water hammer and degradation because of such mechanisms as voided lines. (Note: Steam and water hammers maybe an issue with this design as there may not be high point vents as the loop is under water.) NUREG-0927 provides guidance for steam or water hammer prevention and mitigation and Generic Letter 88-17 provides guidance for shutdown operation.7. Using the system process diagrams, piping and instrumentation diagrams (P&IDs), system descriptions and schematics, failure modes and effects analysis, and component performance specifications, the reviewer determines that the DHR has the heat removal capacity to bring the reactor to a stable condition in a reasonable period of time (with assuming a single failure of an active component with either onsite or off-site electric power available, if this condition is relied upon to perform a safety-related function). For the purposes of this review, the NRC considers 36 hours a reasonable time period. The organization responsible for the review of steam and feedwater systems evaluates the initial cooldown phase for the NuScale iPWR, so this review effort should be coordinated with that reviewer.

8. The staff reviews the cooldown function to determine whether it can be performed from the control room assuming a single failure of an active component, with only either onsite or offsite electric power available. The applicant must justify any operation required outside of the control room. he organization responsible for the review of steam and feedwater systems will evaluate the initial cooldown for the NuScale iPWR.

9. By reviewing the system description and the P&IDs, the reviewer confirms that the DHR system function satisfies the pressure relief requirements of BTP 5-4.

10. By reviewing the piping arrangement and system description of the DHR, the reviewer confirms that the DHR meets the requirements of GDC 5 concerning shared systems.

11. The reviewer of reactor thermal hydraulic systems contacts the reviewer of steam and

feedwater systems in conjunction with the review of the DHR function heat sink interaction to exchange information and ensure that the reviews consider the interfacing parameters consistently. The reviewer of reactor thermal hydraulic systems then evaluates the DHR function description to determine that the DHR design allows for this maximum temperature. Specifically, the reviewer should verify the maximum cooling capacity of the DHRHX using maximum reactor building pool water temperature.

12. The reviewer of reactor thermal hydraulic systems contacts the reviewer of instrumentation and control systems to obtain any needed information from that review. Specifically, the reviewer of instrumentation and control systems confirms that automatic actuation and remote-manual valve controls are capable of performing the functions required, and that sensor and monitoring provisions are adequate. If applicable, t he instrumentation and controls of the DHR must have sufficient redundancy to satisfy the single failure criterion.

13. The reviewer of reactor thermal hydraulic systems contacts the reviewer of containment integrity to exchange information related to their reviews.

14. The reviewer of reactor thermal hydraulic systems contacts the reviewer of the quality assurance and maintenance to discuss any special test requirements and to confirm that the proposed preoperational test program for the DHR function to demonstrate the capability of all systems and components associated with the removal of decay heat.

15. The reviewer evaluates the proposed plant technical specifications as follows:

    A. Confirm the suitability of the limiting conditions of operation, including the proposed time limits and reactor operating restrictions for periods when system equipment is inoperable because of repairs and maintenance.

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

16. The reviewer contacts the reviewers of structural analysis and seismic/geotechnical issues to confirm that the systems employed to remove residual heat are housed in a structure whose design and design criteria provide adequate protection against wind, tornadoes, floods, and missiles, as appropriate.

17. The reviewer provides information to other reviewers in those areas where the organization responsible for the review of reactor thermal hydraulic systems has a review responsibility that is not explicitly covered in steps 1-14 above. These additional areas of review responsibility include:

    A. Identification of engineered safety features and safe-shutdown electrical loads and verification that the minimum time intervals for the connection of the engineered safety features to the standby power systems are satisfactory

    B. Identification of vital auxiliary systems associated with the DHR function and determination of cooling load functional requirements and minimum time intervals

    C. Identification of essential components associated with the main steam supply and

the feedwater system that are required to operate DHR during and following shutdown.

18. The reviewer considers compliance with acceptance criteria requirement II.9 of this document by verifying that a leakage control program includes those portions of the DHR located outside of containment that contains or may contain radioactive material following an accident. The leakage control program should include periodic leak testing and measures to minimize leakage from the DHR.

19. As necessary, the reviewer verifies that actions have been taken to ensure the continued availability and reliability of the decay heat removal systems during shutdown operations.

    For NuScale iPWRs, design features should be incorporated to prevent a loss of DHR functions. The reviewer should verify that the RHR-specific guidance and measures contained in Generic Letter 88-17 and summarized as follows are satisfied:

    A. The reviewer verifies that the applicant/licensee will have measures in place to ensure that the RCS will remain stable and controlled in all operating conditions. These measures include both prevention of a loss of active RHR (e.g., normal main steam and feed system), and enhanced monitoring requirements to ensure timely response to a loss of active RHR, should such a loss occur, and the capability to switch to the passive decay heat removal system.

    B. The reviewer verifies that the applicant/licensee has the capability of continuously monitoring <u>DHR performance and RCS characteristics important for core cooling whenever the active</u> DHR function (i.e., main steam and feed system) is being used for cooling the RCS.

    C. The reviewer verifies that the DHR has visible and audible indications of abnormal conditions in temperature, level, and DHR function performance parameters.

20. The reviewer verifies that new light-water reactor applicants have ensured high reliability of the shutdown decay heat removal system as follows (see SECY-90-016 and the associated staff requirements memorandum (SRM) dated June 26, 1990, SECY-93-087 and the associated SRM dated July 21, 1993, and NUREG-1449):

    A. The reviewer verifies that design provisions exist to reasonably ensure the continuity of flow through the core and DHR.

    B. The reviewer verifies that provisions exist to ensure the availability of reliable systems for decay heat removal.

    C. The reviewer verifies that automatic closure interlocks for the RHR suction isolation valves, if provided, are designed in such a manner as to minimize inadvertent valve closure during system operation.

21. The reviewer verifies that the applicant has reviewed its DHR design configurations to identify any piping connections to the RCS that could be subjected to temperature

distributions that could result in unacceptable thermal stresses. This review should consider the potential for thermal stratification, thermal cycling, and thermal fatigue, given the DHR configuration. The reviewer verifies that appropriate action has been taken, where such piping is identified, to ensure it will not be subjected to unacceptable thermal stresses (see NRC Bulletin 88-08). This review should focus on DHR function configurations; the organization responsible for mechanical engineering reviews under SRP Section 3.9.3 reviews the stress analysis and ensures that it conforms to the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code.

22. Since there is a potential for SG tube leak or tube rupture, the reviewer will verify that, to the extent practical, the NuScale DHR is designed to an ultimate rupture strength at least equal to the normal RCS operating pressure. The NRC states its regulatory position with respect to minimizing the potential for an intersystem LOCA in advanced or evolutionary light-water reactors in SECY-90-016 and SECY-93-087 and their associated SRM. All elements of the DHR are to be considered (e.g., instrument lines, heat exchanger tubes, valve bonnets). The licensee should provide justification for elements not designed to an ultimate rupture strength at least equal to the normal RCS operating pressure.

23. When reviewing the NuScale design, the reviewer should use SECY-94-084 to determine the amount of regulatory oversight necessary. In SECY-94-084, the staff developed new regulatory and review guidance for a reliability assurance program to establish the regulatory treatment of nonsafety systems. The NuScale iPWR advanced light-water reactor (ALWR) design makes extensive use of passive systems to meet regulatory requirements. The NuScale design has used the design philosophy of safety-related passive systems and also the nonsafety-related active systems for RHR. The DHR function is included as a safety-related passive system for use during transients and to provide defense in depth to the non-safety-related systems. The non-safety-related active systems are the first line of defense to reduce challenges to the passive systems in the event of transients or plant upsets. The extensive use of safety-related passive systems and the nonsafety-related active system design philosophy presents a departure from previous licensing practices. (Note: In NuScale, both active and passive methods are used for shutdown, and the active non-safety shutdown system is non-safety grade and hence this section is applicable.)

24. The reviewer verifies that interlocks are in place to prevent DHR from activating with low reactor building pool water level.

25. The reviewer verifies that provisions are provided to clean the heat exchanger fins which are in borated water.

26. 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 DCD meets the acceptance criteria. 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 FSAR.

    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).

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

27. 10 CFR 20.1406(a) requires that the applicant describe how facility 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.

    10 CFR 20.1406(a) applies to this DSRS section because the possibility of leakage of radioactive water into the heat exchanger tubes exists.

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 safety evaluation report. The reviewer also states the bases for those conclusions.

The DHR function for normal plant cooldown is accomplished by first using the main stream and feedwater systems followed by the DHR system and finally by flooding the containment. During a normal shutdown procedure, the steam from the steam generators is routed directly to the condenser via the turbine bypass system. The condensate is pumped back to the steam generators via the feedwater system. At predetermined conditions the system transitions to decay heat removal by the passive, closed-loop DHR system. When conditions permit the containment is flooded and decay heat is removed primarily by conduction.

During transient conditions, when the normal secondary side cooling is unavailable, the DHR system actuates to transfer heat from the RCS to the reactor building pool. Since the DHR has no direct contact with the RCS, lowering the RCS pressure and temperature for actuation of the DHR function is not required. The DHR valves upstream of the DHRHX are opened and actuate the decay heat removal function via natural circulation around the DHR loop. The DHR acts like an auxiliary feedwater system, removing residual decay heat from the reactor core via the steam generators and transferring that heat to the UHS (i.e., the reactor building pool) via the DHRHXs. The scope of review of the NuScale DHR for the plant function included piping and instrumentation diagrams or equivalent figures, system descriptions and schematics, equipment layout drawings, failure modes and effects analysis, and design performance specifications for essential risk-significant components. The review included the applicant's proposed design criteria and design bases for the DHR function, analysis of the adequacy of those criteria and bases, and conformance of the design to those criteria and bases.

The staff concludes that the design of the DHR function is acceptable and meets the requirements of GDCs 1, 2, 4, 5, 14, 19, 34, 44, 45, 46, 54, 57, 10 CFR 50.62, 10 CFR 50.63,10 CFR 50.34(f)(2)(xxvi) and 10 CFR 20.1406. This conclusion is based on the following:

A. The applicant has met GDC 1 requirements for the DHR function. Acceptance is based on the structures, systems, and components important to safety as being designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Recognized codes and standards shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product. Appropriate records of the design, fabrication, erection, and testing of structures, systems, and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.

B. The applicant has met GDC 2 requirements with respect to Regulatory Position C-2 of RG 1.29 concerning the seismic design of SSCs whose failure could cause an unacceptable reduction in the capability of the DHR.

C. The applicant has met GDC 4 requirements with respect to dynamic effects associated with flow instabilities and loads (e.g., water hammer).

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

E. The applicant has met the requirements of GDC 14 with respect to reactor coolant pressure boundary design, fabrication, erection and testing so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure and of gross rupture.

F. The applicant has met GDC 19, with respect to the main control room requirements for normal operations and shutdown, and GDC 34, which specifies requirements for the DHR by meeting the regulatory positions in BTP 5-4.

G. The applicant has met GDC 34 requirements for RHR. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor at a rate such that specified acceptable fuel design limits and the design of the reactor coolant pressure boundary are not exceeded. The applicant has demonstrated that the DHR can perform its function of RHR under transient conditions. The applicant has demonstrated that the system has suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities to assure that the safety function can be accomplished, assuming a single failure concurrent with a loss of either onsite or

offsite power . The applicant has demonstrated that the DHR in conjunction with safety related emergency core cooling can perform the function of RHR during accident conditions. The design has been evaluated for adequate margins related to heat exchanger heat removal performance during normal and accident conditions.

H. The applicant has met GDC 44 requirements for a system capable of transferring heat loads from SSCs important to safety to an ultimate heat sink under normal operating and accident conditions. The system shall have suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities to assure that for only either onsite electric power operation or offsite electric power operation, the system safety function can be accomplished, assuming a single failure. The applicant has demonstrated that the DHR is capable of transferring heat generated in the reactor core to the reactor building pool via the DHRHXs at a rate sufficient enough to ensure reactor safety and integrity during normal operating and accident conditions. The applicant has demonstrated that suitable redundancy of the DHR components and features is sufficient to assure the safety function can be accomplished with or without a coincident loss of offsite power, concurrent with a single failure.

I. The applicant has met GDC 45 requirements for design provisions that allow for appropriate periodic inspection of important components of the DHR, such as heat exchangers and piping, to assure system integrity and capability. The applicant has demonstrated that important DHR components can be periodically inspected to assess system degradation and/or incipient failure so that a high level of assurance that the DHR will function reliably can be maintained.

J. The applicant has met GDC 46 requirements for the DHR being designed for appropriate periodic pressure and functional testing of the structural and leak-tight integrity of its components, the operability and performance of the system active components, and the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation for reactor shutdown including operation of portions of the protection system and transfer between normal and emergency power sources. The applicant has demonstrated that a high level of assurance can be maintained in regards to the DHR reliability based on periodic testing to detect degradation in system performance or in the system pressure boundary.

K. The applicant has met GDC 54 requirements, which provides reasonable assurance that the containment isolation system will isolate piping systems penetrating containment reliably as required.

L. The applicant has met GDC 57 requirements for automatic containment isolation for those portions of the system which penetrate primary reactor containment and are not part of the reactor coolant pressure boundary.

M. The applicant has met the parameters in Item III.D.1.1 of NUREG-0737, equivalent to 10 CFR 50.34(f)(2)(xxvi) for applicants subject to 10 CFR 50.34(f), with respect

to leakage detection and control in the design of DHR systems outside containment that contain (or may contain) radioactive material following an accident.

N. The applicant has met the requirements of 10 CFR 50.62. Acceptance is based on the DHR design providing for automatic initiation of the auxiliary feedwater-like function under conditions indicative of an ATWS.

O. The applicant has met the requirements of 10 CFR 50.63. Acceptance is based on the DHR design providing for sufficient decay heat removal in a station blackout in accordance with Positions 3.2.2, 3.3.2, and 3.3.4 of RG 1.155.

P. The applicant meets 10 CFR 20.1406 requirements for minimization of contamination of the facility and the environment, and for avoiding design features that would interfere with eventual decommissioning.

V. IMPLEMENTATION

The regulations in 10 CFR 52.17(a)(1)(xii), 10 CFR 52.47(a)(9), and 10 CFR 52.79(a)(41) establish requirements for applications for ESPs, DCs, and COLs, respectively. These regulations require the application to include an evaluation of the site (ESP), standard plant design (DC), or facility (COL) against the Standard Review Plan (SRP) revision in effect six months before the docket date of the application. While the SRP provides generic guidance, the staff developed the SRP guidance based on the staff's experience in reviewing applications for construction permits and operating licenses for large light-water nuclear power reactors. The proposed small modular reactor (SMR) designs, however, differ significantly from large light-water nuclear reactor power plant designs.

In view of the differences between the designs of SMRs and the designs of large light-water power reactors, the Commission issued SRM- COMGBJ-10-0004/COMGEA-10-0001, "Use of Risk Insights to Enhance the Safety Focus of Small Modular Reactor Reviews," dated August 31, 2010 (ML102510405) (SRM). In the SRM, the Commission directed the staff to develop risk-informed licensing review plans for each of the SMR design reviews, including plans for the associated pre-application activities. Accordingly, the staff has developed the content of the DSRS as an alternative method for the evaluation of a NuScale-specific application submitted pursuant to 10 CFR Part 52, and the staff has determined that each application may address the DSRS in lieu of addressing the SRP, with specified exceptions. These exceptions include particular review areas in which the DSRS directs reviewers to consult the SRP and others in which the SRP is used for the review. If an applicant chooses to address the DSRS, the application should identify and describe all differences between the design features (DC and COL applications only), analytical techniques, and procedural measures proposed in an application and the guidance of the applicable DSRS section (or SRP section as specified in the DSRS), and discuss how the proposed alternative provides an acceptable method of complying with the regulations that underlie the DSRS acceptance criteria.

The staff has accepted the content of the DSRS as an alternative method for evaluating whether an application complies with NRC regulations for NuScale SMR applications, provided that the application does not deviate significantly from the design and siting assumptions made by the NRC staff while preparing the DSRS. If the design or siting assumptions in a NuScale application deviate significantly from the design and siting assumptions the staff used in preparing the DSRS, the staff will use the more general guidance in the SRP as specified in 10 CFR 52.17(a)(1)(xii), 10 CFR 52.47(a)(9), or 10 CFR 52.79(a)(41), depending on the type of application. Alternatively, the staff may supplement the DSRS section by adding appropriate criteria in order to address new design or siting assumptions.

VI. REFERENCES

1. 10 CFR 50.34(f), "Additional TMI-Related Requirements."

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

3. 10 CFR Part 50, Appendix A, GDC 4, "Environmental and Dynamic Effects Design Bases."

4. 10 CFR Part 50, Appendix A, GDC 5, "Sharing of Structures, Systems and Components."

5. 10 CFR Part 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary"

6. 10 CFR Part 50, Appendix A, GDC 19, "Control Room."

7. 10 CFR Part 50, Appendix A, GDC 34, "Residual Heat Removal."

8. 10 CFR Part 50, Appendix A, GDC 54, "Primary Systems Penetrating Containment"

9. BTP 5-4, "Design Requirements of the Residual Heat Removal System."

10. RG 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident."

11. RG 1.29, "Seismic Design Classification."

12. RG 1.68, "Initial Test Programs for Water-Cooled Nuclear Power Plants."

13. RG 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants."

14. Deleted

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

16. RG 1.215, “Guidance for ITAAC Closure under 10 CFR Part 52.”

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

18. SRM, “SECY 90-016 - Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationships to Current Regulatory Requirements,” June 26, 1990.

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

20. SRM, “SECY 93-087 ’Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs,” July 21, 1993.

21. Generic Letter 88-17, “Loss of Decay Heat Removal” October 17, 1988.

22. Generic Letter 89-04, “Guidance on Developing Acceptable Inservice Testing Programs,” April 3, 1989.

23. Generic Letter 90-06, “Resolution of Generic Issue 70, ‘Power-Operated Relief-Valve and Block Valve Reliability,’ and Generic Issue 94, ‘Additional Low-Temperature Overpressure Protection for Light-Water Reactors,” June 25, 1990.

24. Generic Letter 92-02, “Resolution of Generic Issue 79, ‘Unanalyzed Reactor Vessel (PWR) Thermal Stress During Natural Convection Cooldown’, March 6, 1992.

25. Deleted

26. Deleted

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

28. NUREG-0660, “NRC Action Plan Developed as a Result of the TMI-2 Accident.”

29. NUREG-0718, “Licensing Requirements for Pending Applications for Construction Permits and Manufacturing License.”

30. NUREG-0737, “Clarification of TMI Action Plan Requirements.”

31. NUREG-1316, “Technical Findings and Regulatory Analysis Related to Generic Issue 70 - Evaluation of Power-Operated Relief Valve and Block Valve Reliability in PWR Nuclear Power Plants.”

32. NUREG-1449, “Shutdown and Low-Power Operation at Nuclear Power Plants in the United States.”

33. NUREG-0927, Revision 1, “Evaluation of Water Hammer Occurrences in Nuclear Power

Plants," March 1984.

34. SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs," March 28, 1994.

35. SECY-95-132, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems (RTNSS) in Passive Plant Designs," May 22, 1995.