U.S. NUCLEAR REGULATORY COMMISSION

DESIGN-SPECIFIC REVIEW STANDARD for NuScale SMR DESIGN

BRANCH TECHNICAL POSITION 5-4

DESIGN REQUIREMENTS OF THE DECAY HEAT REMOVAL (DHR) SYSTEM RESPONSIBILITIES

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

Secondary - None

The NuScale integral pressurized water small modular reactor (SMR) makes extensive use of passive systems to meet regulatory requirements. Residual heat removal (RHR) for the NuScale SMR during normal shutdown is not only provided by the main condenser and feedwater systems, but it is also provided by flooding of the primary containment. The decay heat removal (DHR) system, a passively engineered safety feature, provides secondary-side cooling for non-loss-of-coolant accident design-basis events when normal secondary-side cooling is unavailable. The DHR system function is included as a safety-related passive system that provides a similar function as the auxiliary feedwater system at a typical, large pressurized-water reactor (PWR) in the event of a loss of main feedwater. The NuScale nonsafety-related RHR function is provided by the turbine bypass system, feedwater system in conjunction with the primary containment flooding. In some situations, the DHR system works alongside the NuScale emergency core cooling system (ECCS) feature in order to remove core decay heat.

A. BACKGROUND

General Design Criterion (GDC) 19 in Appendix A to Part 50 of the *Code of Federal Regulations* (10 CFR Part 50) states, “A control room shall be provided from which actions can be taken to operate the nuclear power unit under normal conditions...”

Normal operating conditions include the shutting down of the reactor; therefore, since the DHR system is one of several systems involved in the shutdown of the reactor, this system must be operable from the control room.

GDC 34, “Residual Heat Removal,” states, “Suitable redundancy shall be provided to assure that for onsite electrical power system operation (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.”

In most current PWR plant designs, the RHR system is in the primary side, is an active system with a lower design pressure than the reactor coolant system (RCS), is located outside of

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containment, and is part of the ECCS. However, the NuScale safety-related DHR system is in the secondary side and is a passive design with different design characteristics.

The NuScale safety-related DHR system is capable of operating without operator intervention, does not interface directly with the reactor coolant system, and is located outside of containment where the DHR heat exchanger is submerged in the reactor pool. Although the NuScale DHR system is not a direct part of the NuScale ECCS, the two heat removal systems can operate simultaneously providing a diverse means of RHR if necessary.

B. BRANCH TECHNICAL POSITION

1. Functional Requirements for DHR

The system(s) that can be used to take the reactor from normal operating conditions to stable shutdown shall satisfy the following functional requirements:

A. The design shall be such that the reactor can be taken from normal operating conditions to safe/stable shutdown. These systems shall satisfy GDC 1 through 5. For NuScale, the safety-related systems are passive; however, the active backup systems may be risk significant but nonsafety-related as determined under Standard Review Plan (SRP) Section 19.0 using the selection criteria under Section C.IV.9.3 of Regulatory Guide (RG) 1.206, “Combined License Applications for Nuclear Power Plants.”

B. The passive safety-related system(s) shall have suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities to ensure that, for onsite electrical power system operation (assuming offsite power is not available) and offsite electrical power system operation (assuming onsite power is not available), the system function can be accomplished assuming a single failure (electrical as well as mechanical).

C. The passive safety-related system shall be capable of being operated or controlled from the control room (including instrumentation for monitoring and control functions) with either only onsite or offsite power available. In demonstrating that the passive safety-related system can perform its function assuming a single failure, limited operator action outside of the control room would be considered acceptable if suitably justified.

D. The passive safety-related system(s) shall be capable of bringing the reactor to a safe/stable shutdown condition, with only offsite or onsite power available, within a reasonable period of time following shutdown, assuming the most limiting single failure.

2. Pressure Relief Requirements for DHR

The DHR system shall be designed to withstand the highest anticipated transient pressure condition (e.g., steam generator tube rupture) consistent with the appropriate margins as specified by the American Society of Mechanical Engineers Boiler and Pressure Vessel Code.

3. Test Requirements

The isolation valve operability and interlock circuits must be designed so as to permit online testing when operating in the DHR mode. Testability shall meet the requirements of Institute of Electrical and Electronics Engineers (IEEE) Standard 338-1987 and RG 1.22.

The pre-operational and initial startup test program shall be in conformance with RG 1.68, "Initial Test Programs for Water-Cooled Nuclear Power Plants." The programs for NuScale shall include tests with supporting analysis to confirm that (1) adequate mixing of borated water added before or during cooldown can be achieved under natural circulation conditions in the NuScale design where chemical shim is utilized and permit estimation of the times required to achieve such mixing and (2) cooldown under natural circulation conditions can be achieved within the limits specified in the emergency operating procedures.

4. Operational Procedures

The operational procedures for bringing the plant from normal operating power to safe/stable shutdown shall be in conformance with RG 1.33, "Quality Assurance Program Requirements (Operation)." For NuScale, the operational procedures shall include specific procedures and information required for cooldown under natural circulation conditions.

5. 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 design-specific review standard (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.

C. REFERENCES

1. 10 CFR Part 50, Appendix A, GDC 1, “Quality Standards and Records.”

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

3. 10 CFR Part 50, Appendix A, GDC 3, “Fire Protection.”

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

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

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

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

8. U.S. Nuclear Regulatory Commission, “Periodic Testing of Protection System Actuation Functions,” Regulatory Guide 1.22 (ADAMS Accession No. ML083300530).

9. U.S. Nuclear Regulatory Commission, “Quality Assurance Program Requirements (Operation),” Regulatory Guide 1.33 (ADAMS Accession No. ML003739995).

10. U.S. Nuclear Regulatory Commission, “Initial Test Programs for Water-Cooled Nuclear Power Plants,” Regulatory Guide 1.68 (ADAMS Accession No. ML13051A027).

11. U.S. Nuclear Regulatory Commission, “Initial Test Programs for Water-Cooled Nuclear Power Plants,” Regulatory Guide 1.68 (ADAMS Accession No. ML13051A027).

12. U.S. Nuclear Regulatory Commission, “Loss of Decay Heat Removal—10 CFR 50.54(f),” Generic Letter 88-17, October 17, 1988.

13. U.S. Nuclear Regulatory Commission, “Resolution of Generic Issue 70, ‘Power-Operated Relief Valve and Block Valve Reliability’ and Generic Issue 94, ‘Additional Low-Temperature Over Pressure Protection for Light-Water Reactors’ pursuant to 10 CFR 50.54(f),” Generic Letter 90-06, June 25, 1990.

14. U.S. Nuclear Regulatory Commission, “Resolution of Generic Issue 79, ‘Unanalyzed Reactor Vessel (PWR) Thermal Stress during Natural Convection Cooldown,’” Generic Letter 92-02, March 6, 1992.

15. U.S. Nuclear Regulatory Commission, “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,” NUREG-1316.

16. Institute of Electrical and Electronics Engineers, IEEE Standard 338-1987, “IEEE Standard Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems.”