U.S. NUCLEAR REGULATORY COMMISSION

DESIGN- SPECIFIC REVIEW STANDARD FOR NuScale SMR DESIGN

6.2.4 CONTAINMENT ISOLATION SYSTEM

Primary - Organization responsible for the review of containment integrity

Secondary - None

I. AREAS OF REVIEW

NuScale is an integral, pressurized water, small modular reactor (SMR) with the reactor, steam generator, pressurizer, and control rod drives all located in a single pressure vessel. The NuScale reactor containment is an evacuated, low alloy steel vessel surrounding the smaller reactor vessel and immersed in a bay of a large pool containing borated water inside the reactor building that serves as the passive ultimate heat sink for containment heat removal.

The containment isolation system allows the normal or emergency passage of fluids through the containment boundary while preserving the ability of the boundary to prevent or limit the escape of fission products from postulated accidents. This Design Specific Review Standard (DSRS) section, therefore, addresses the isolation of fluid systems penetrating the containment boundary, including design and testing requirements for isolation barriers and actuators. Isolation barriers include valves, closed piping systems, and blind flanges.

The NuScale containment isolation system includes the following classifications of equipment:

1. Safety-related and risk-significant equipment
2. Safety-related and nonrisk-significant equipment
3. Nonsafety-related and risk-significant Regulatory Treatment of Nonsafety Systems (RTNSS) equipment
4. Nonsafety-related nonrisk-significant equipment.

The NuScale application will include the classification of systems, structures, and components (SSCs), a list of risk significant SSCs, and a list of RTNSS equipment. Based on this information, the staff will review according to SRP Sections 3.2.1, 3.2.2, 17.4 and 19.3 to confirm the determination of safety-related and risk-significant SSCs.

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The specific areas of review are as follows:

1. The design of containment isolation provisions, including:

    A. The number and location of isolation valves (i.e., the isolation valve arrangements and their physical locations as to the containment).

    B. The actuation and control features for isolation valves.

    C. The positions of isolation valves for normal plant operating conditions (including shutdown), post-accident conditions, and valve operator power failures.

    D. The valve actuation signals.

    E. The basis for selection of closure times of isolation valves.

    F. The mechanical redundancy of isolation devices.

    G. The acceptability of closed piping systems inside containment as isolation barriers.

2. The protection of containment isolation provisions against loss of function from missiles, pipe whip, and earthquakes.

3. The environmental conditions inside and outside the containment considered in the design of isolation barriers.

4. The design criteria applied to isolation barriers and piping.

5. The provisions for detecting needs to isolate remote manual-controlled systems like engineered safety feature systems.

6. The design provisions and Technical Specifications (TS) for testing of isolation barrier operability and leakage rate.

7. The calculation of containment atmosphere released prior to isolation valve closure for lines that provide direct paths to the environs.

8. The containment purging/evacuation/venting design features minimizing purging/evacuating time consistent with as low as reasonably achievable principles for occupational exposure.

9 The reliability of the purge or evacuation system in isolating under accident conditions.

10. The containment isolation and valve indication provisions for station blackout (SBO).

11. <u>Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC)</u>. 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 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.

12. 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

Other SRP and DSRS sections interface with this section as follows:

1. SRP Sections 3.2.1 and 3.2.2: review of system seismic design and quality group classification, respectively.

2. SRP Section 3.6.2: review of postulated pipe rupture locations, the containment penetration exclusion area, and related dynamic effects on containment isolation capability.

3. DSRS Section 3.8.2: review of the containment isolation system structural design for adequate protection against earthquakes.

4. SRP Section 3.9.2: review of the containment isolation system mechanical design for adequate protection against breach of integrity, missiles, pipe whip, and jet impingement.

5. SRP Section 3.10: evaluation of the qualification test program for electric valve operators and the operability assurance program for containment purge, evacuation and vent valves.

6. SRP Section 3.10 and DSRS Section 3.11: review of sensing and actuation instrumentation of the plant protection system located both inside and outside of containment.

7. DSRS Section 7.1: evaluation of the actuation and control features for isolation valves.

8. DSRS Sections 8.3.1 and 8.3.2: review of the power sources for containment isolation valve operators in each line penetrating the containment for whether any single fault can prevent isolation of the line.

9. DSRS Section 8.4: review of capability to withstand or cope with and recover from SBO coordinated with review of containment isolation system for appropriate system functioning for SBO.

10. DSRS Section 15.6.5: review of the closure time for containment isolation valves in lines that provide a direct path to the environs for the prediction of onset of accident-induced fuel failure.

11. DSRS Section 14.2: review of verification programs including the Initial Plant Test

Program.

12. DSRS Section 15.0.3: review of the radiological dose consequence analysis for the release of containment atmosphere prior to closure of containment isolation valves in lines providing a direct path to the environs.

13. DSRS Section 16.0: review at the operating license stage of proposed TS for operability and leakage-rate testing of isolation barriers and closure time for containment isolation valves.

14. DSRS Section 16.0 and SRP Section 19: evaluation of the risk significance of the actuation and control features and closure time for containment isolation valves, as well as the equipment survivability of the containment penetrations in a beyond design basis accident addressed in SRP section 19.2.

15. SRP Section 17.4 evaluation of the Reliability Assurance Program.

16. SRP Sections 17.6 and 13.4 evaluation of the Maintenance Rule Program.

The specific acceptance criteria and review procedures are contained in the referenced DSRS sections.

II. ACCEPTANCE CRITERIA

Requirements

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

1. General Design Criterion (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 16, Containment Design.

6. GDC 54, Piping Systems Penetrating Containment.

7. GDCs 55 and 56, Reactor Coolant Pressure Boundary Penetrating Containment, as to isolation valves for lines penetrating (GDC 55) the primary containment boundary as parts of the reactor coolant pressure boundary or as direct connections to the containment atmosphere (GDC 56) as follows:

    A. One locked-closed isolation valve[1] inside and one outside containment; or

    B. One automatic isolation valve inside and one locked-closed isolation valve

[1] Locked-closed isolation valves are defined as sealed closed barriers (see DSRS Acceptance Criteria II.6).

outside containment; or

C. One locked-closed isolation valve inside and one automatic isolation valve[2] outside containment; or

D. One automatic isolation valve inside and one outside[2] containment.

8. GDC 57, as it relates to the requirement that lines penetrating the primary containment boundary and neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere have at least one locked-closed, remote-manual, or automatic isolation valve[2] outside containment.

9. 10 CFR 50.34(f)(2)(xiv) Provide containment isolation systems that: (II.E.4.2)

   A. Ensure all non-essential systems are isolated automatically by the containment isolation system.

   B. each non-essential penetration (except instrument lines) have two isolation barriers in series.

   C. Do not result in the reopening of the containment isolation valves on resetting of the isolation signal.

   D. Utilize a containment setpoint pressure for initiating containment isolation as low as is compatible with normal operation.

   E. Include automatic closing on a high radiation signal for all systems that provide a path to the environs.

10. 10 CFR 50.34(f)(2)(xv) Provide a capability for containment purging/venting designed to minimize the purging time consistent with ALARA principles for occupational exposure, and that provide and demonstrate high assurance that the purge system will reliably isolate under accident conditions. (II.E.4.4)

11. 10 CFR 50.34(f)(2)(xix) Provide instrumentation adequate for monitoring plant conditions following an accident that includes core damage. (II.F.3)

12. 10 CFR 50.34(f)(3)(iv) Provide one or more dedicated containment penetrations, equivalent in size to a single 3-foot diameter opening, in order not to preclude future installation of systems to prevent containment failure, such as a filtered vented containment system. (II.B.8)

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 facility that incorporates the design certification has been constructed and will be operated in

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[2] A simple check valve is not an acceptable automatic isolation valve for use outside containment.

conformity with the design certification, the provisions of the Atomic Energy Act (AEA), and the U.S. Nuclear Regulatory Commission's (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 combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.

14. 10 CFR 52.47(a)(8) and 10 CFR 52.79(a)(17), as they relate to demonstrating compliance with any technically relevant portions of the Three Mile Island (TMI)-related requirements set forth in 10 CFR 50.34(f)(2)(xiv) and 10 CFR 50.34(f)(2)(xv), for DC and COL reviews, respectively.

15. 10 CFR 50.63(a)(2), as it relates to ensuring that appropriate containment integrity is maintained in the event of a station blackout for a specified duration.

These GDCs establish requirements for the design, testing, and functional performance of isolation barriers in lines penetrating the primary containment boundary and, in general, require two isolation valves in series to maintain the isolation function, assuming any single, active failure in the containment isolation provisions. However, containment isolation provisions different from the explicit requirements of GDCs 55 and 56 are acceptable if the differences are justified.

<u>DSRS Acceptance Criteria</u>

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

1. Regulatory Guide (RG) 1.11 describes acceptable containment isolation provisions for instrument lines. In addition, instrument lines closed both inside and outside containment are designed to withstand pressure and temperature conditions following a loss-of-coolant accident (LOCA) and dynamic effects are acceptable without isolation valves.

2. Containment isolation provisions for lines in engineered safety feature or engineered safety feature-related systems may include remote-manual valves, but should detect possible leakage from these lines outside containment.

3. Containment isolation provisions for lines in systems needed for safe shutdown of the plant (e.g., liquid poison system, reactor core isolation cooling system, and isolation condenser system) may include remote-manual valves, but there should be provisions for detecting leakage from such lines outside containment.

4. Containment isolation provisions for lines in the systems of items 2 and 3 normally

consist of one isolation valve inside and one outside containment. If it is not practical to locate a valve inside containment (for example, the valve may be under water as a result of an accident), both valves may be located outside containment. For this type of isolation valve arrangement, the valve nearer the containment and the piping between the containment and the valve should be enclosed in a leak-tight or controlled-leakage housing. If, in lieu of housing, the piping and valve are designed to preclude a breach of piping integrity, the design should comply with SRP Section 3.6.2 requirements. Design of the valve or the piping compartment should provide the capability to detect and terminate leakage from the valve shaft or bonnet seals.

5. Containment isolation provisions for lines in engineered safety feature or engineered safety feature-related systems normally consist of two isolation valves in series. A single isolation valve is acceptable if system reliability can be shown to be greater, the system is closed outside containment, and a single active failure can be accommodated with only one isolation valve in the line. The closed system outside containment should be protected from missiles, designed to seismic Category I and Group B quality standards, and have a design temperature and pressure rating at least equal to that for the containment. The closed system outside containment should be leak-tested unless system integrity can be shown to be maintained during normal plant operations. For this type of isolation valve arrangement the valve is located outside containment, and the piping between the containment and the valve should be enclosed in leak-tight or controlled-leakage housing. If, in lieu of housing, piping and valve are designed conservatively to preclude a breach of piping integrity, the design should comply with SRP Section 3.6.2 requirements. Design of the valve or the piping compartment should provide the capability to detect and terminate leakage from the valve shaft or bonnet seals.

6. Sealed-closed barriers may be used in place of automatic isolation valves. Sealed-closed barriers include blind flanges and sealed-closed isolation valves which may be closed manual valves, closed remote-manual valves, or closed automatic valves which remain closed after a LOCA. Sealed-closed isolation valves should be under administrative control so they cannot be opened inadvertently. Administrative control includes mechanical devices to seal or lock the valve closed or to prevent power supply to the valve operator.

7. Relief valves may be used as isolation valves provided the relief setpoint is greater than 1.5 times the containment design pressure.

8. 10 CFR 50.34(f)(2)(xiv) requires that systems penetrating the containment be classified as either essential or nonessential. Reference 33 presents guidance on the classification of systems as essential and nonessential. Essential systems, like those described in items 2 and 3, may include remote-manual containment isolation valves, but there should be provisions for detecting leakage from the lines outside containment. 10 CFR 50.34(f)(2)(xiv) also requires that nonessential systems be isolated automatically by the containment isolation signal.

9. Isolation valves outside containment should be located as close to it as practical, as required by GDCs 55, 56, and 57.

10. To meet the requirements of GDCs 55 and 56, upon loss of actuating power, automatic isolation valves should take the position of greatest safety. The position of an isolation

valve for normal and shutdown plant operating and post-accident conditions depends on the fluid system function. If a fluid system has no post-accident function, the isolation valves in the lines should be closed automatically. For engineered safety feature or engineered safety feature-related systems, isolation valves in the lines may remain open or be opened. In a power failure to the valve operator isolation valves should be in the "safe" position, normally the post-accident valve position. For lines equipped with motor-operated valves, a loss of actuating power leaves the affected valve in the "as-is" position, which may be the open position; however, redundant isolation barriers ensure that the isolation function for the line is satisfied. All power-operated isolation valves should have position indications in the main control room.

11. To improve the reliability of the isolation function, addressed in GDC 54, 10 CFR 50.34(f)(2)(xiv) requires reduction of the containment setpoint pressure that initiates containment isolation for nonessential penetrations to the minimum value compatible with normal operating conditions.

12. There should be diversity in the parameters sensed for the initiation of containment isolation to satisfy the GDC 54 requirement for reliable isolation capability.

13. To improve the reliability of the isolation function, addressed in GDC 56, system lines which provide open paths from the containment to the environs (e.g., purge, evacuation and vent lines addressed in 10 CFR 50.34(f)(2)(xiv)) should be equipped with radiation monitors capable of isolating these lines upon a high-radiation signal, which should not be considered a diverse containment isolation parameter.

14. In meeting GDC 54 requirements, the performance capability of the isolation function should reflect the safety importance of isolating system lines. Consequently, containment isolation valve closure times should be selected for rapid isolation of the containment following postulated accidents. Valve closure time for a power-operated valve to be in the fully-closed position after the actuator power has reached the operator assembly does not include the time to reach actuation signal setpoints or instrument delay times, which, with system design capabilities, should be considered for establishing valve closure times. For lines providing open paths from the containment to the environs (e.g., the containment purge or evacuation and vent lines), isolation valve closure times of five seconds or less may be necessary. The closure times of these valves should be established to minimize the release of containment atmosphere to the environs, to mitigate the offsite radiological consequences, and to prevent degradation of emergency core cooling system effectiveness by reduced containment back-pressure. Analyses of the radiological consequences and the effect on the containment back-pressure of the release of containment atmosphere should justify the selected valve closure time. Branch Technical Position (BTP) 6-4 for iPWRs presents additional guidance on the design and use of containment purge or evacuation systems which may be used during the normal plant operating modes (i.e., startup, power operation, hot standby, and hot shutdown).

    Containment purge or evacuation valves that do not satisfy the operability criteria of Branch Technical Position 6-4 must be sealed closed as defined in subsection II.6 of this DSRS section during operational conditions 1, 2, 3, and 4. Furthermore, closure of these valves must be verified at least every 31 days. These requirements should be incorporated into the TS for plant operation.

15. The use of a closed system inside containment as one of the isolation barriers is

acceptable if the closed system design satisfies the following requirements:

A. The system does not connect with either the reactor coolant system or the containment atmosphere.

B. The system is protected against missiles and pipe whip.

C. The system is designated seismic Category I.

D. The system is classified Quality Group B.

E. The system is designed to withstand temperatures equal to at least that of the containment design.

F. The system is designed to withstand the external pressure from the containment structure acceptance test.

G. The system is designed to withstand the LOCA transient and environment.

As to the structural design of containment internal structures and piping systems, the protection against loss of function from missiles, pipe whip, and earthquakes is acceptable if 1) isolation barriers are located behind missile barriers; 2) pipe whip was considered in the design of pipe restraints and the location of piping penetrating the containment; and 3) the isolation barriers, including the piping between isolation valves, are designated seismic Category I, i.e., designed to withstand the effects of the safe-shutdown earthquake, as recommended by Regulatory Guide 1.29.

16. To meet the requirements of GDCs 1, 2, 4, and 54, appropriate reliability and performance considerations should be included in the design of isolation barriers to reflect the safety importance of their integrity (i.e., containment capability) under accident conditions. The design criteria for components performing a containment isolation function, including the isolation barriers and the piping between them or the piping between the containment and the outermost isolation barrier, are acceptable if:

    A. Group B quality standards, as defined in RG 1.26, apply to the components, unless the service function dictates that Group A quality standards apply.

    B. The components are designated seismic Category I in accordance with RG 1.29.

17. GDC 54 requires reliable isolation capability; therefore, for remote-manual isolation valves, the design of the containment isolation system is acceptable if there are provisions to allow the operator in the main control room to know when to isolate fluid systems equipped with remote-manual isolation valves. Such provisions may include instruments to measure flow rate, sump water level, temperature, pressure, and radiation level.

18. GDC 54 specifies requirements for the containment isolation system; therefore, to satisfy GDC 54, the design of the containment isolation system should provide for operability testing of the containment isolation valves and leakage rate testing of the isolation barriers. The isolation valve testing program should be consistent with that proposed for

other engineered safety features. DSRS Section 6.2.6 presents acceptance criteria for the leakage rate testing program for containment isolation barriers.

19. GDC 54 requires reliable isolation capability. To satisfy this requirement, the design of the containment isolation system should reduce the possibility of unintended isolation valve reopening following isolation. 10 CFR 50.34(f)(2)(xiv) requires control systems for automatic containment isolation valves be designed for resetting the isolation signal without automatically reopening the valves. Reopening of containment isolation valves should require deliberate operator action and combined reopening of containment isolation valves is not acceptable. Reopening of isolation valves must be valve by valve or line by line, provided that electrical independence and other single-failure criteria remain satisfied. Administrative provisions to close all isolation valves manually before resetting the isolation signals is not an acceptable method for meeting this design requirement.

20. In meeting 10 CFR 50.34(f)(2)(xv) purging requirements, the regulatory guidance of BTP 6-4, "Containment Purging During Normal Plant Operations," should be used to establish compliance with this regulation.

21. RG 1.155, "Station Blackout," Regulatory Position C.3.2.7, provides guidance for meeting the requirements of the SBO rule, 10 CFR 50.63(a)(2), for containment isolation valves and valve position indication.

22. 10 CFR Part 50, Appendix K, provides guidance for the determination of the extent of fuel failure (source term) in the radiological calculations.

Technical Rationale

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

1. GDC 1, "Quality Standards and Records," requires that safety-related SSCs be designed, fabricated, erected, and tested to quality standards commensurate with the safety functions performed.

   This DSRS section defines appropriate reliability and performance standards for the design of the containment isolation system. These standards reflect the importance of forming an essentially leak-tight barrier to prevent the release of fission products in an accident. RG 1.26 specifies quality standards applicable to containment isolation system components. This DSRS section also contains TMI-related requirements for containment isolation dependability, containment purging/evacuating/venting during plant operation, and purge/evacuation/vent valves.

   Compliance with GDC 1 requirements provides reasonable assurance that the containment isolation system will perform its safety function and prevent the release of radioactive materials to the environment.

2. GDC 2, "Design Bases for Protection Against Natural Phenomena," requires that safety-related SSCs be designed to withstand the effects of natural phenomena like earthquake, tornado, hurricane, flooding, tsunami, and seiche without loss of capability to perform safety functions.

   GDC 2 applies to this DSRS section because the reviewer evaluates the containment isolation system for its capability to isolate the containment under accident conditions (e.g., LOCA) combined with severe natural phenomena. RG 1.29 provides guidance acceptable to the staff for developing designs with the capability to withstand earthquakes.

Compliance with GDC 2 requirements provides reasonable assurance that the containment will act as an essentially leak-tight barrier and prevent the release of radioactive materials to the environment under all plausible conditions.

3. GDC 4, "Environmental and Dynamic Effects Design Bases," requires that safety-related SSCs (A) be designed to accommodate the effects of, and be compatible with, environmental conditions of normal operation, maintenance, testing, and postulated accidents (including LOCAs) and (B) be protected appropriately against dynamic effects (including those of missiles, pipe whipping, and discharging fluids) of equipment failures and events and conditions outside the nuclear power unit.

   GDC 4 applies to this DSRS section because the reviewer evaluates the containment isolation system for its capability to perform its isolation function at all times in any environmental condition to which the system's components may be exposed, including dynamic effects. BTP 6-4 provides guidance as to dynamic effects that should be considered in the design of containment purge or evacuation and vent valves.

   Compliance with GDC 4 requirements provides reasonable assurance that the containment isolation system has the capability to perform its safety function of containment isolation and to prevent the release of radioactive materials to the environment. These requirements also ensure that containment purge or evacuation and vent valves are designed for reliable isolation under accident conditions.

4. GDC 5, “Sharing of Structures Systems and Components” requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair the ability to perform their safety functions including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

   With regard to the containment isolation system, GDC 5 requires the component part of the containment isolation system be essentially independent in order to ensure that an accident in one unit of a multiple-unit facility will not propagate to other units. Therefore the containment isolation system, including the associated instrumentation and controls for each unit should be designed for independence from other units.

   GDC 5 requirements provide assurance that a failure or accident in one unit will not affect additional units of a multi-unit site.

5. GDC 16, "Containment Design," requires that the reactor containment and its systems establish an essentially leak-tight barrier against the uncontrolled release of radioactive materials to the environment.

   GDC 16 applies to this DSRS section because the reviewer evaluates the containment isolation system for whether it allows the normal or emergency passage of fluids through the containment boundary while preserving the capability of the boundary to prevent or limit the escape of fission products from postulated accidents. This DSRS section provides guidance as to design requirements for containment isolation provisions, including the number and location of isolation valves, their actuation and control features, redundancy, valve actuation signals, and closure times.

Compliance with GDC 16 requirements provides reasonable assurance that the containment and its systems will act as an essentially leak-tight barrier to prevent the uncontrolled release of radioactive materials to the environment in an accident.

6. GDC 54, "Piping Systems Penetrating Containment," requires that piping systems that penetrate the primary reactor containment have leak-detection, isolation, containment, redundancy, reliability, and performance capabilities that reflect the safety importance of isolating these piping systems.

   GDC 54 applies to this DSRS section because the reviewer evaluates the containment isolation system for whether valves in piping systems that penetrate the containment are designed to close reliably under accident conditions and prevent the uncontrolled release of radioactive materials. To ensure reliability of these valves, this DSRS section provides guidance as to leak detection, redundancy, leakage testing, and functional testing. RGs 1.11 and 1.141 provide guidance acceptable to the staff for isolating instrument lines that penetrate the containment and for fluid systems, respectively. Nonessential lines are isolated automatically by the containment isolation signal.

   Compliance with GDC 54 requirements provides reasonable assurance that the containment isolation system will isolate piping systems penetrating containment reliably as required.

7. GDC 55, "Reactor Coolant Pressure Boundary Penetrating Containment," requires that each line of the reactor coolant pressure boundary penetrating the primary reactor containment meet specified criteria for the use and positioning of isolation valves.

   GDC 55 applies to this DSRS section because the reviewer evaluates the containment isolation system to ensure that there is no direct connection between the primary coolant and the plant environs. This assurance is provided by specific requirements for isolation valves (i.e., locked-closed, automatic, or a combination of locked-closed and automatic) on both sides of the containment barrier. Isolation valves outside the containment should be located as close to the containment as is practical. Upon loss of actuating power, automatic valves must take the position of greatest safety. Other requirements (e.g., those for higher quality in design, additional inservice inspection, and protection against severe natural phenomena) may be imposed based on use and physical characteristics of the plant-site environs.

   Compliance with GDC 55 requirements provides reasonable assurance that lines penetrating the containment and connected to the reactor coolant system will not be sources of excessive offsite radiation doses due to either line rupture or failure of a valve to close.

9. GDC 56, "Primary Containment Isolation," requires that each line that connects directly to the containment atmosphere and penetrates the primary reactor containment must meet specified criteria for the use and positioning of isolation valves.GDC 56 applies to this DSRS section because the reviewer evaluates the containment isolation system to ensure that (A) there is no direct connection between the containment atmosphere and

the plant environs or (B) if there is direct communication (as that during containment purging, evacuation or venting) that the lines can be reliably isolated. This assurance is provided by specific requirements for isolation valves (i.e., locked-closed, automatic, or a combination of locked-closed and automatic) on both sides of the containment barrier. Isolation valves outside the containment should be located as close to the containment as is practical. Upon loss of actuating power, automatic valves must take the position of greatest safety. BTP 6-4 contains specific requirements for containment purge or evacuation and vent valves, providing a high degree of assurance that these valves will isolate reliably under accident conditions.

Compliance with GDC 56 requirements provides reasonable assurance that lines penetrating the containment and connected to the containment atmosphere will not be sources of excessive offsite radiation doses due to either line rupture or failure of a valve to close.

10. GDC 57, "Closed System Isolation Valves," requires, for each line penetrating the primary reactor containment which is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere, at least one containment isolation valve that is automatic, locked-closed, or capable of remote-manual operation. Isolation valves must be located on the outside of the containment barrier as close to the containment as is practical.

    GDC 57 applies to this DSRS section because the reviewer evaluates the containment isolation system for whether there is no direct connection between the fluids in the closed system and the plant environment. Assurance is by specific requirements for a closed system and for an isolation valve that is locked-closed, automatic, or capable of remote-manual operation. A single valve is specified because the system is closed; hence, failure of the valve to close would not, by itself, allow contact between fluids in the closed system and the plant environment.

    Compliance with GDC 57 requirements provides reasonable assurance that lines penetrating the containment and connected to closed systems will not be sources of excessive offsite radiation doses due to line rupture or failure of a valve to close.

11. 10 CFR 50.63 requires that all light-water-cooled nuclear power plants be able to withstand and recover from an SBO, that necessary systems be capable of cooling the core, and that appropriate containment integrity be maintained in SBO. Guidance for compliance with 10 CFR 50.63 is provided in RG 1.155. As many safety systems necessary to support safe operation and shutdown of the reactor depend on alternating current (AC) power, the consequences of an SBO could be severe, particularly if the integrity of barriers to prevent the release of fission products (e.g., fuel cladding, reactor coolant pressure boundary, containment) are not maintained throughout the event and its recovery period. The containment isolation system, including its provisions for control, indication, and performance under loss/restoration of power conditions, is instrumental in maintaining integrity of the containment barrier without undue interference with flow paths essential for cooling the reactor core. Compliance with 10 CFR 50.63 and the positions of RG 1.155 in the performance of the containment isolation system for an SBO, therefore adds defense in depth against unacceptable offsite radiological consequences if both offsite and onsite emergency AC power systems fail by maintaining containment integrity for such an event

III. REVIEW PROCEDURES

The reviewer will select material from the procedures described below, as may be appropriate for a particular case.

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.

The procedures provide guidance on review of the containment isolation system. Portions of the review may be done generically for aspects of containment isolation common to a class of containments or by adoption of the results of previous reviews of plants with essentially the same containment isolation provisions.

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. The reviewer determines acceptability of the containment isolation system by comparison of the system design criteria to the design requirements for an engineered safety feature. The quality standards and the seismic design classification of the containment isolation provisions, including the piping penetrating the containment, are compared to RGs 1.26 and 1.29, respectively.

   The reviewer also ascertains whether any single fault can prevent isolation of the containment by reviewing the containment isolation provisions for each line penetrating the containment for two isolation barriers in series and by reviewing the power sources to the valve operators.

   The technical information in the applicant's submittal justifying containment isolation provisions which differ from the explicit requirements of GDCs 55, 56, and 57 is reviewed. The acceptability of these containment isolation provisions is based on a comparison to the acceptance criteria of subsection II of this DSRS section.

   The isolation valve positions are reviewed for normal and shutdown plant operating conditions, post-accident conditions, and valve operator power failure conditions as listed in the applicant's technical submittal. The position of an isolation valve for each condition depends on the system function. Power-operated valves in fluid systems having no post-accident safety function (nonessential systems) should close automatically.

   In the event of a power failure to a valve operator, the valve position should be that of greater safety, normally the post-accident position; however, special cases are considered individually for the acceptability of the prescribed valve positions. The reviewer also ascertains from the applicant's technical submittal whether all power-operated isolation valves have position indicators in the main control room.

4. Reviewers responsible for the structural design of the containment internal structures and piping systems, including restraints, ensure that the containment isolation provisions are protected adequately against missiles, pipe whip, and earthquakes. The review determines whether for all containment isolation provisions, missile protection and

protection against loss of function from pipe whip and earthquakes were design considerations. The system drawings (which should show the locations of missile barriers as to the containment isolation provisions) are reviewed for whether isolation provisions are protected from missiles. The design criteria for the containment isolation provisions are reviewed for whether protection against dynamic effects like pipe whip and earthquakes was considered in the design. The reviewer requests review of the design adequacy of piping and valves for which conservative design in lieu of leak-tight housing is assumed to preclude possible breach of system integrity.

5. The signals from the plant protection system to initiate containment isolation are reviewed. In general, there should be a diversity of parameters sensed (e.g., abnormal conditions in the reactor coolant system, the secondary coolant system, and the containment) generating containment isolation signals. As plant designs differ and many different signal combinations from the plant protection system initiate containment isolation, the reviewer considers proposed arrangements individually for overall acceptability of the containment isolation signals. The containment setpoint pressure that initiates containment isolation for nonessential penetrations is reviewed. This pressure setpoint should be the minimum value compatible with normal operation, as required by 10 CFR 50.34(f)(2)(xiv)(D). Additional guidance for review of this setpoint is presented in Item II.E.4.2 of NUREG-0737.

6. The reviewer verifies that the control system for automatic containment isolation valves is designed for resetting of the isolation signal without automatic reopening of containment isolation valves and that combined reopening of isolation valves is not possible.

7. Systems having post-accident safety functions (essential systems) may have remote-manual isolation valves in the lines penetrating the containment. Provisions for detecting leakage from these lines outside containment and for allowing the operator in the main control room to isolate the system train if leakage occurs are reviewed. Leakage detection provisions may include instrumentation for measuring system flow rates or the pressure, temperature, radiation, or water level in areas outside the containment like valve rooms or engineered safeguards areas. Acceptance of the leakage detection provisions described in the applicant's technical submittal is based on capability to detect leakage and identify lines that should be isolated. The reviewer determines whether the containment isolation provisions are designed for individual leak-testing of isolation barriers. This information should be tabulated in the applicant's technical submittal to facilitate review.

8. The reviewer determines from the applicant's descriptive technical information whether provisions in the design of the containment isolation system allow periodic operability testing of the power-operated isolation valves and the containment isolation system. At the COL stage of review, the reviewer determines whether the content and intent of proposed TS for operability and leak-testing of containment isolation equipment agree with requirements developed by the staff. In particular, there should be the following technical specifications: containment purge, evacuation or vent valves that do not satisfy BTP 6-4 operability criteria must be sealed closed as defined in subsection II.6 (Acceptance Criteria) of this DSRS section and verified sealed closed at least every 31 days during all operational conditions except cold shutdown; refueling; purging, evacuating or venting time should be minimized consistent with as low as reasonably achieved principles for occupational exposure and 10 CFR 20.1406 requirements for minimizing contamination; and containment purge, evacuation or vent valves with

resilient seals must be subjected to leakage-testing and periodic resilient seal replacement.

9. The reviewer determines the acceptability of the use of closed systems inside containment as isolation barriers by comparing the system designs to the Acceptance Criteria of Subsection II of this DSRS section.

10. Isolation valve closure times are reviewed. In general, valve closure times should be less than one minute regardless of valve size (See the Acceptance Criteria for valve closure times in Subsection II of this DSRS section). Valves in lines that provide direct paths to the environs (e.g., the containment purge or evacuation) may have to close in times much shorter than one minute. Closure times for these valves may be dictated by radiological dose analyses or emergency core cooling system performance considerations. The reviewer requests reviews of analyses justifying closure times for these valves as necessary.

11. The reviewer evaluates the design features of the purging/evacuating/venting system for minimizing purging/evacuating time and verifies whether there is a high degree of assurance that the purge/evacuation system will isolate reliably under accident conditions.

12. The reviewer verifies whether appropriate containment integrity is maintained in SBO by the capability, independent of the preferred and blacked-out unit's onsite emergency AC power supplies, for valve position indication and closure for containment isolation valves that may be in pen positions at the onset of SBO. Certain containment isolation valves are excluded from consideration as addressed in RG 1.155.

13. 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 applicant's design control document (DCD). The reviewer should also consider the appropriateness of identified COL action items. The reviewer may identify additional COL action items; however, to ensure these COL action items are addressed during a COL application, they should be added to the 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).

    For review of both DC and COL applications, DSRS 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.

IV. EVALUATION FINDINGS

The reviewer verifies that the applicant has provided sufficient information and that the review and calculations (if applicable), 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 staff concludes that the containment isolation system functional design is acceptable and meets the requirements of GDCs 1, 2, 4, 16, 54, 55, 56, and 57, 10 CFR Part 50, Appendix K, the additional TMI-related requirements 10 CFR 50.34(f)(2)(xiv) and 10 CFR 50.34(f)(2)(xv), and the SBO requirements of 10 CFR 50.63(a)(2). The conclusion is based on the following findings:

1. The applicant has met the requirements of [regulation] for [limits of review under regulation] by (for each item applicable to the review how it was met and why acceptable for the regulation):

   A. Meeting the regulatory positions in NUREG __________ or RGs _____;

   B. Meeting an alternative method to regulatory positions in RG _____ reviewed by the staff and found acceptable;

   C. Meeting the regulatory position in BTP ________;

   D. Using calculation methods (for what was evaluated) previously reviewed by the staff and found acceptable; the staff has reviewed the impact parameters in this case and found them suitably conservative or has performed independent calculations to verify acceptability of their analysis; or

   E. Meeting the provisions (industry standard number and title) reviewed by the staff and determined to be appropriate for this application.

2. Repeat discussion for each regulation cited.

   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 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," subparagraph (2)(xiv), regarding TMI Action Plan Item II.E.4.2, "Containment Isolation Dependability."

2. 10 CFR 50.34(f), "Additional TMI-Related Requirements," subparagraph (2)(xv), regarding TMI Action Plan Item II.E.4.4, "Purging."

3. 10 CFR 50.34(f), "Additional TMI-Related Requirements," subparagraph (2)(xix), regarding TMI Action Plan Item II.F.3, “Instruments for Monitoring Accident Conditions (regulatory guide 1.97)”

4 10 CFR 50.34(f),"Additional TMI-Related Requirements," subparagraph (f)(3)(iv), regarding TMI Action Plan Item II.B.8, “Dedicated Containment Penetration”

5. 10 CFR 50.63, "Loss of All Alternating Current Power," subparagraph (a)(2), regarding containment integrity in the event of a station blackout.

6. 10 CFR Part 50, Appendix A, GDC 1, "Quality Standards and Records."

7. 10 CFR Part 50, Appendix A, GDC 2, "Design Bases for Protection Against Natural Phenomena."

8. 10 CFR Part 50, Appendix A, GDC 4, "Environmental and Missile Design Basis."

9 10 CFR Part 50, Appendix A, GDC 5, “Sharing of Structures, Systems, and Components.”

10. 10 CFR Part 50, Appendix A, GDC 16, "Containment Design."

11. 10 CFR Part 50, Appendix A, GDC 54, "Piping Systems Penetrating Containment."

12. 10 CFR Part 50, Appendix A, GDC 55, "Reactor Coolant Pressure Boundary Penetrating

Containment."

13. 10 CFR Part 50, Appendix A, GDC 56, "Primary Containment Isolation."

14. 10 CFR Part 50, Appendix A, GDC 57, "Closed System Isolation Valves."

15. 10 CFR Part 50, Appendix K, "ECCS Evaluation Models."

16. 10 CFR Part 100, "Reactor Site Criteria."

17. RG 1.11, "Instrument Lines Penetrating Primary Reactor Containment."

18. RG 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants."

19. RG 1.29, "Seismic Design Classification."

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

21. RG 1.141, "Containment Isolation Provisions for Fluid Systems."

22. RG 1.155, "Station Blackout."

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

24. RG 1.182, “Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants.”

25. RG 1,206, “Combined License Applications for Nuclear Power Plants (LWR Edition).”

26. RG 1.215, “Guidance for ITAAC Closure Under 10 CFR Part 52.26."

27. BTP 6-4, "Containment Purging During Normal Plant Operations."

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

29. NUREG-0718, "Licensing Requirements for Pending Application for Construction Permits and Manufacturing License."

30. NUREG-1449, "Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States," Final Report, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, September 1993.

31. NRC Generic Letter 83-02, "NUREG-0737 Technical Specifications," January 10, 1983.

32. NRC Letter to all Holders of Operating Licenses and Construction Permits for Pressurized Water Reactors (PWRs), “Loss of Decay Heat Removal (Generic Letter 88-17),” October 17, 1988.

33. Item II.E.4.2, "Containment Isolation Dependability," in NUREG-0737 and NUREG-0718.