Draft for Comment

U.S. NUCLEAR REGULATORY COMMISSION

DESIGN-SPECIFIC REVIEW STANDARD FOR NuScale SMR DESIGN

9.3.2 PROCESS AND POST ACCIDENT SAMPLING SYSTEMS

REVIEW RESPONSIBILITIES

Primary - Organization responsible for the review of the functionality and capability of the process sampling systems (PSSs)

Secondary - Organization responsible for the review of radiation protection

Organization responsible for the review of chemical engineering issues

Organization responsible for the review of thermal-hydraulic performance of safety systems in pressurized water reactors and boiling water reactors

Organization responsible for the review of systems associated with the balance of plant

Organization responsible for the review of instrumentation and control systems

Organization responsible for the review of emergency planning

I. AREAS OF REVIEW

This review standard addresses the design and performance of the PSSs. PSSs have the capability to sample all normal process systems and principal components, including provisions for obtaining samples at various physical points in the system and times during operation.

This Design-Specific Review Standard (DSRS) section is applicable to design certification (DC) combined license (COL), and early site permits (ESP) applications submitted under Title 10 of *Code of Federal Regulations* (10 CFR) Part 52.

The specific areas of review are as follows:

1. The design objectives and design criteria for the PSSs are reviewed to confirm the design and evaluate the adequacy of the applicant's Technical Specifications in these areas. The review includes identification of the process streams to be sampled and the parameters to be determined through sampling.

2. The system descriptions for the PSSs are reviewed. The review includes (a) system description and schematics or piping and instrumentation diagrams (P&IDs), if applicable, (b) provisions for obtaining representative samples, (c) the location of sampling points and sample stations, and (d) provisions for purging sampling lines.

ADAMS Accession No. ML15131A298

3. The seismic design and quality group classifications of piping and equipment, and the bases for the classifications chosen are reviewed, including the expected temperatures and pressures and the materials of construction of system components.

4. The isolation provisions for the systems and the means provided to limit radioactive releases by minimizing reactor coolant losses are reviewed.

5. The PSSs' operational procedures for sampling the reactor coolant and containment atmosphere to determine the capability to promptly obtain samples for chemical and radiochemical analyses are reviewed.

6. The Post Accident Sampling System (PASS) is not mandatory. PASS was eliminated in some designs (NRC letters to Combustion Engineering Owners Group and Westinghouse Owners group) because their PSSs' designs allow for the collection and analysis of: (1) highly radioactive samples for containment pH and (2) containment atmosphere for hydrogen and other fission products, such that there will be no decrease in the effectiveness of emergency plans in the absence of a dedicated PASS. In lieu of the PASS, the following actions are necessary for the applicant to qualify process sampling for taking radioactive samples without having a specific post-accident sampling capability:

   A. Establish the capability for classifying a fuel damage event at the Alert level threshold

   B. Develop contingency plans for obtaining highly radioactive samples of the reactor coolant, containment sump, and containment atmosphere

   C. Determine for its own plant(s) that no decrease in the effectiveness of emergency plans will result from not having post-accident sampling system capability

   D. Establish the capability to sample and analyze hydrogen in the containment atmosphere (recommended)

   E. Maintain offsite capability to monitor radioactivity, including radioactive iodines

7. <u>Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC)</u>. For DC and 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.

8. <u>COL Action Items and Certification Requirements and Restrictions</u>. 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 Standard Review Plan (SRP) and DSRS sections interface with this section as follows:

1. SRP Sections 3.2.1 ”Seismic Classification” and 3.2.2 “System Quality Group Classification”: review of the acceptability of the seismic and quality group classifications for system components.

2. DSRS Sections 3.3.1 ”Wind Loadings,” 3.3.2 “Tornado Loadings,” 3.4.2 “Analysis Procedures,” 3.5.3 “Barrier Design Procedures,” 3.7.1 “Seismic Design Parameters,” 3.7.2 “Seismic System Analysis,” 3.7.3 “Seismic Subsystem Analysis,” 3.7.4 “Seismic Instrumentation,” 3.8.4 “Other Seismic Category I Structures,” and 3.8.5 “Foundations”: review of the acceptability of the design analyses, 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 the safe shutdown earthquake, the probable maximum flood, and tornadoes and tornado missiles.

3. DSRS Section 3.4.1 “Internal Flood Protection for Onsite Equipment Failures”: review of internal flood protection for piping, tank, and vessel failures, operation of the fire protection system, and water leakage into safety-related areas.

4. DSRS Sections 3.5.1.1, “Internally Generated Missiles (Outside Containment)”, 3.5.1.2, Internally Generated Missiles (Inside Containment), 3.5.1.3, Turbine Missiles, and 3.5.2 ”Structures, Systems, and Components to be Protected from Externally-Generated Missiles,” and SRP 3.5.1.6 “Aircraft Hazards”: review of the protection against internally and externally generated missiles.

5. SRP Section 3.6.1 “Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment”: review of the design with respect to the effects of externally or internally generated missiles, pipe whip, and jet impingement forces associated with postulated pipe breaks in high-energy fluid systems or leakage cracks in moderate-energy fluid systems.

6. SRP Section 3.6.2 “Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping”: review of possible break locations in high- and moderate-energy systems during normal plant operation and the review of dynamic effects (e.g., pipe whip, jet impingement) of pipe breaks.

7. SRP Section 3.10 “Seismic and Dynamic Qualification of Mechanical and Electrical Equipment”: review of the seismic qualification of Category I instrumentation.

8. DSRS Section 3.11 “Environmental Qualification of Mechanical and Electrical Equipment”: verification that those valves that are inaccessible during an accident are environmentally qualified to ensure operability under accident conditions.

9. SRP and DSRS Sections in 6.2 “Containment Design”: review of the environmental effects of piping failures inside containment.

10. DSRS Section 6.2.4 “Containment Isolation System”: verification that remotely operated containment isolation valves in the PSSs are designed to close on a containment isolation signal or safety injection signal.

11. DSRS Section 6.6 “Inservice Inspection and Testing of Class 2 and 3 Components”: verification that in-service inspection requirements are met for system components.

12. DSRS Chapter 7 “Instrumentation and Controls”: verification that override capability exists for containment isolation valves that will be used for process sampling of the reactor coolant, containment sump water, and containment atmosphere without clearing the containment isolation signal or safety injection signal. Review of the types of instruments needed for flood protection, including the adequacy of detectors and alarms necessary to detect rising water levels within structures, and the review of the consequences of flooding on other safety-related instrumentation and electrical equipment.

13. DSRS Section 8.3.1 “A-C Power Systems (Onsite)”: verification that power supplies are available to all remotely operated containment isolation valves in the sampling system, after detection of an accident that requires containment isolation and assuming a concurrent loss of offsite power.

14. DSRS Section 11.2 “Liquid Waste Management System”: review of liquid radwaste processing.

15. DSRS Section 11.3 “Gaseous Waste Management System”: review of the ventilation systems that may operate during sampling of radioactive materials.

16. DSRS Section 11.5 “Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems”: review of the sampling and monitoring systems for radwaste processing systems.

17. DSRS Sections 12.3 -12.4 “Radiation Protection Design Features”: review of potential personal radiation exposure during process sampling of radioactive material.

18. SRP Section 13.3 “Emergency Planning”: review of contingency plans and procedures for post-accident sampling.

19. DSRS Section 16.0 “Technical Specifications”: review of Technical Specifications and short-term availability controls for post-accident sampling.

20. SRP Chapter 17 “Quality Assurance”: review of quality assurance.

21. Determination of PSS safety-related or risk-significant functions and review of regulatory treatment of nonsafety systems (RTNSS) are coordinated and performed under SRP Chapter 19 “Severe Accidents.”

II. ACCEPTANCE CRITERIA

Requirements

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

1. 10 CFR 20.1101(b), as it relates to providing engineering controls based upon sound radiation protection principles to achieve occupational doses and doses to members of the public as low as is reasonably achievable (ALARA).

2. 10 CFR 20.1406, which requires the applicant to describe how facility design will minimize, to the extent practical, contamination of the facility and the environment, generation of radioactive waste, and facilitate eventual decommissioning.

3. General Design Criterion (GDC) 1, found in Appendix A to 10 CFR Part 50, as it relates to the design of the PSSs and components in accordance with standards commensurate with the importance of their safety functions.

4. GDC 2, as it relates to the ability of the PSSs to withstand the effects of natural phenomena.

5. GDC 13, as it relates to monitoring variables that can affect the fission process, the integrity of the reactor core, and the reactor coolant pressure boundary.

6. GDC 14, as it relates to assuring the integrity of the reactor coolant pressure boundary by sampling for chemical species that can affect the reactor coolant pressure boundary.

7. GDC 26, as it relates to reliably controlling the rate of reactivity changes by sampling boron concentration.

8. GDC 41, as it relates to reducing the concentration and quality of fission products released to the environment and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents.

9. GDC 60, as it relates to the capability of the PSSs to control the release of radioactive materials to the environment.

10. GDC 63, as it relates to detecting conditions that may result in excessive radiation levels in the fuel storage and radioactive waste systems.

11. GDC 64, as it relates to monitoring the containment atmosphere and plant environs for radioactivity.

12. Three Mile Island (TMI) Action Plan Item III.D.1.1 in NUREG-0737, as it relates to the provisions for a leakage control program to minimize the leakage from those portions of the PSSs outside of the containment that contain or may contain radioactive material following an accident. Requirements regarding applicability of 10 CFR 50.34(f) are provided in 10 CFR 52.47(a)(8) for DC reviews and 10 CFR 52.79(a)(17) for COL reviews. 10 CFR 50.34(f)(2)(xxvi) provides leakage control and detection requirements for those applicants subject to 10 CFR 50.34(f).

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 operate in 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 AEA, and the NRC's regulations.

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.

1. The applicant's design is such that the PSSs have the capability to sample all normal process systems and principal components, including provisions for obtaining samples from at least the points indicated below. The guidelines of Regulatory Guide (RG) 1.21, Position C.2 and the Electric Power Research Institute (EPRI) PWR Water Chemistry Guidelines are used to meet the requirements of the relevant GDC.

For NuScale	GDC
Reactor coolant (e.g., letdown system)	13, 14, 26, 64, 60
Refueling water storage tank	13
Boric acid mix tank	13, 26
Emergency Boration System boron injection tank	13
Chemical additive tank (if applicable)	13, 14, 41
Spent fuel pool	63, 60
Secondary coolant (e.g., condensate hotwell)	13, 14
Steam generator blowdown (if applicable)	14, 64, 60
Secondary coolant condensate treatment waste	64, 60

For NuScale	GDC
Sumps inside containment	64, 60
Containment atmosphere	64, 60

DSRS Section 11.5 gives other sample points that may be included in the PSSs but do not require remote sampling.

2. 10 CFR 20.1406. Minimization of contamination to the facility and the environment, and designs to facilitate eventual decommissioning, will be considered acceptable if the design identifies provisions to detect contamination that may enter as inleakage from other systems, identifies potential collection points such as water treatment systems or system low points, and addresses the long term control of radioactive material in the system. DC/COL-ISG-06 and RG 4.21 relate to acceptable levels of detail and content required to demonstrate compliance with 10 CFR 20.1406.

3. The plant Technical Specifications include the required analysis and frequencies.

4. The following guidelines should be used to determine the acceptability of the PSSs' functional designs:

   A. Provisions should be made to ensure representative samples from liquid process streams and tanks. For tanks, provisions should be made to sample the bulk volume of the tank and to avoid sampling from low points or from potential sediment traps. For process stream samples, sample points should be located in turbulent flow zones. The guidelines of Regulatory Position C.2 in RG 1.21 are followed to meet these criteria.

   B. Provisions should be made to ensure representative samples from gaseous process streams and tanks in accordance with American National Standards Institute/Health Physics Society (ANSI/HPS) Standard N13.1-1999. The guidelines of Regulatory Position C.2.3 in RG 1.21 are followed to meet this criterion.

   C. Provisions should be made for purging sampling lines and for reducing plateout in sample lines (e.g., heat tracing). The guidelines of Regulatory Position C.2.5 in RG 1.21 are followed to meet this criterion.

   D. Provisions should be made to purge and drain sample streams back to the system of origin or to an appropriate waste treatment system in accordance with the requirements of 10 CFR 20.1101(b) to keep radiation exposures at ALARA levels. The guidelines of Regulatory Positions 2.d.(2), 2.f.(3), and 2.f.(8) in RG 8.8 are followed to meet this criterion.

   E. Isolation valves should fail in the closed position, in accordance with the requirements of GDC 60 to control the release of radioactive materials to the environment.

   F. Passive flow restrictions to limit reactor coolant loss from a rupture of the sample line should be provided in accordance with the requirements of

10 CFR 20.1101(b) to keep radiation exposures to ALARA levels and the requirements of GDC 60 to control the release of radioactive materials to the environment. The guidelines of Regulatory Position 2.i.(6) in RG 8.8 should be followed to meet this criterion. Redundant environmentally qualified, remotely operated isolation valves may replace passive flow restrictions in the sample lines to limit potential leakage. The automatic containment isolation valves should close on containment isolation signals or safety injection signals.

5. To meet the requirements of GDCs 1 and 2, the applicant's seismic design and quality group classification of sampling lines, components, and instruments for the PSSs should conform to the classification of the system to which each sampling line and component is connected (e.g., a sampling line connected to a Quality Group A and seismic Category I system should be designed to Quality Group A and seismic Category I classification), in accordance with Regulatory Positions C.1, C.2, and C.3 in RG 1.26; Regulatory Positions C.1, C.2, C.3, and C.4 in RG 1.29, and the guidelines of RG 1.97. Components and piping downstream of the second isolation valve may be designed to Quality Group D and nonseismic Category I requirements, in accordance with Regulatory Position C.3 in RG 1.26.

6. 10 CFR 52.47(a)(1)(vi) specifies that the application of a design certification should contain proposed ITAAC for SSCs necessary and sufficient to assure the plant is built and will operate in accordance with the design certification. 10 CFR 52.80(a) specifies that the COL identifies the ITAAC for SSCs necessary and sufficient to assure that the facility has been constructed and will be operated in conformity with the license. DSRS Section 14.3 provides guidance for reviewing the ITAAC. The requirements of 10 CFR 52.47(a)(1)(vi) and 10 CFR 52.80(a) will be met, in part, by identifying inspections, tests, analyses, and acceptance criteria of the top-level design features of the PSSs in the design certification application and the combined license, respectively.

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. 10 CFR 20.1101(b) requires that licensees use, to the extent practicable, procedures and engineering controls based upon sound radiation protection principles to achieve doses that are ALARA. The radiation protection community has recognized that it is prudent to avoid unnecessary exposure to radiation and to maintain doses at ALARA levels based on the assumption that a nonthreshold linear relationship exists between dose and biological effects that is independent of the dose rate. The objective of the ALARA requirement for the PSSs are to ensure that licensees make every reasonable effort in planning, design, and operation of the system to maintain exposures to radiation as far below the limits of 10 CFR Part 20 as is reasonably achievable. Appropriate station layout and design features should be provided to reduce the potential doses to personnel who must operate, service, or inspect the station PSSs. The safety benefit of implementing radiation protection goals for the PSSs is to reduce doses wherever and whenever reasonably achievable, thereby reducing the risk that is assumed (for radiation protection purposes) to be proportional to the dose.

2. 10 CFR 20.1406 requires the design of a nuclear power unit to address minimization of contamination of the facility and the environment, and ease of eventual

decommissioning. 10 CFR 20.1406 applies to this DSRS section because the PSSs will connect with contaminated systems. DC/COL-ISG-06 and RG 4.21 provide guidance to meet 10 CFR 20.1406. Specific guidance to meet 10 CFR 20.1406 is identified in RG 4.21 Positions C.1 through C.4.

3. 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 PSSs are important to safety in that (a) through connections to systems, such as the reactor coolant system, that are designed and classified in accordance with recognized quality standards, their failure could adversely affect the integrity of these systems, (b) during normal operation, the PSSs provide information that allows the operator to assess the integrity of the fuel cladding and to recognize conditions that could jeopardize the integrity of the reactor coolant pressure boundary, (c) although the PSSs do not have a specific post-accident sampling capability, its design allows for the collection and analysis of highly radioactive samples that may be present following an accident. Meeting the requirements of GDC 1 ensures that the PSSs will be designed, fabricated, erected, and tested to generally accepted and recognized codes and standards that are sufficient to ensure a quality system in keeping with the required safety functions.

4. GDC 2 requires that SSCs important to safety be designed to withstand the effects of natural phenomena without the loss of the capability to perform their safety functions. The PSSs connect to systems, such as the reactor coolant pressure boundary, that are designed to seismic Category I requirements. Those portions of the PSSs or components that form interfaces between seismic Category I and nonseismic Category I features should be designed to seismic Category I requirements. Meeting the requirements of GDC 2 for those portions of the PSSs that interface with seismic Category I systems will enhance plant safety by ensuring the integrity of seismic Category I systems, such as the reactor coolant pressure boundary, during the design-basis seismic event.

5. GDC 13 requires that instrumentation be provided to monitor variables and systems to ensure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, and the reactor coolant pressure boundary. The PSSs are relied upon to provide water and gaseous samples from the reactor coolant system and associated auxiliary systems during all normal modes of operation. Satisfying the requirements of GDC 13 for the PSSs ensures that important information is provided for evaluating whether safety systems and other systems important to safety are performing their intended safety functions (i.e., reactivity control, fuel cladding integrity, maintaining reactor coolant system integrity, and maintaining containment integrity).

6. GDC 14 requires that the reactor coolant pressure boundary be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, rapidly propagating failure, and gross rupture. The PSSs are relied upon during normal operating, transient, and postulated accident conditions to provide primary and secondary water chemistry data. Verification that key chemistry parameters, such as chloride, hydrogen, and oxygen concentrations, are within prescribed limits and that impurities are properly controlled provides assurance that the many mechanisms for corrosive attack will be mitigated and will not adversely affect the reactor coolant pressure boundary. Minimizing the potential for corrosive chemical attack increases

plant safety by decreasing the probability that the reactor coolant pressure boundary will be compromised because of degradation from corrosive chemical attack.

7. GDC 26 establishes requirements regarding the reliable control of the rate of reactivity changes . Meeting the requirements of GDC 26, as it relates to the sampling systems, ensures sampling and evaluation of boron concentrations in the emergency boration injection system tanks and the RCS following an emergency boration, enhancing plant safety by (a) meeting the combined reactivity control system redundancy and capability requirements and (b) not exceeding acceptable fuel design limits.

8. GDC 41 requires that systems to control fission products, hydrogen, oxygen, and other substances that may be released into the reactor containment be provided as necessary to reduce the concentration and quality of fission products released to the environment following postulated accidents. Meeting the requirements of GDC 41, as it relates to the PSS, ensures that sufficient sample information can be provided to verify the safety function of engineered safety features to reduce the concentration and quality of fission products that may be released to the environment following postulated accidents.

9. GDC 60 requires that means be provided to control the release of radioactive materials to the environment. Meeting the requirements of GDC 60, as it relates to the PSSs, enhances safety by providing a means in the design to control the release of radioactive material to the environment. Application of GDC 60 provides reasonable assurance that the PSSs are designed, constructed, installed, and operated on a level commensurate with the need to protect the health and safety of the public and plant operating personnel.

10. GDC 63 requires that systems be provided to monitor the fuel storage and radioactive waste systems to detect conditions that may result in excessive radiation levels. Meeting the requirements of GDC 63, as it relates to the PSSs, ensures that sampling methods are available to monitor the spent fuel pool and the gaseous radwaste storage tank radioactivity levels such that personnel exposures are maintained at ALARA levels.

11. GDC 64 requires that means be available for monitoring the containment atmosphere, spaces after a loss-of-coolant accident, effluent discharge paths, and the plant environs for radioactivity that may be released during normal operations, anticipated operational occurrences, and postulated accidents. Meeting the requirements of GDC 64, as it relates to the PSSs, ensures that a means is provided to monitor the release of radioactive materials, giving the plant operator the indications needed to initiate actions when necessary to protect the health and safety of plant personnel and the general public.

12. TMI Action Plan Item III.D.1.1 in NUREG-0737 requires a program and provisions for leakage control and detection for systems outside containment that contain (or might contain) source term radioactive materials following an accident. The PSSs provide a means for sampling the reactor coolant systems and containment atmosphere to provide the information necessary to assess and control the plant under accident conditions. Because this system draw samples directly from the reactor coolant systems or from the containment atmosphere, it has the potential for containing source term radioactive material during the course of an accident. To prevent unnecessarily high exposures to workers and the public and to maintain control and use of the systems during an accident, a program should be implemented to minimize leakage from this system to as

low as practical levels. Meeting the guidance of TMI Action Plan Item III.D.1.1 in NUREG-0737, or 10 CFR 50.34(f)(2)(xxvi) for those applicants subject to 10 CFR 50.34(f), as they relate to the PSSs, enhances safety by minimizing the leakage from these systems and thereby minimizing the potential exposures to workers and the public, and by providing reasonable assurance that excessive leakage will not prevent the use of the systems under accident conditions.

III. REVIEW PROCEDURES

These review procedures are based on the identified DSRS acceptance criteria. For deviations from these acceptance criteria, the staff should review the applicant's evaluation of how the proposed alternatives provide an acceptable method of complying with the relevant NRC requirements identified in Subsection II.

1. Selected Programs and Guidance - In accordance with the guidance in NUREG-0800, "Introduction - Part 2: Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: 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. In the review of the PSSs, the primary review organization and the organization responsible for the review of chemical engineering issues review the process sampling points as shown on schematics or P&IDs, if applicable, and compare the list of process sampling points contained in the safety analysis report with the sampling points identified in Item 1 of the DSRS Acceptance Criteria, above, to ensure that the required process sampling points have been provided.

4. The primary review organization and the organization responsible for the review of radiation protection compare the capability of the system to obtain representative samples of process fluids and the locations of sampling points with the guidelines for obtaining representative samples of fluids contained in Regulatory Position C.2 of RG 1.21 and with the principles for obtaining representative samples of gases contained in ANSI/HPS N13.1-1999.

5. The primary review organization and the organization responsible for the review of radiation protection verify that provisions made for purging sampling lines and for reducing plateout in sample lines (e.g., heat tracing) conform with the guidelines of Regulatory Position C.2.5 in RG 1.21.

6. The primary review organization, the organization responsible for the review of systems associated with the balance of plant, and the organization responsible for the review of radiation protection verify that provisions have been made to purge and drain sample streams back to the system of origin or to an appropriate waste treatment system to keep radiation exposures at ALARA levels, and that these provisions conform with the guidelines of Regulatory Positions 2.d.(2), 2.f.(3), and 2.f.(8) in RG 8.8.

7. The PSSs are reviewed to ensure that they meet the requirements of 10 CFR 20.1406 for which guidance is provided in DC/COL-ISG-06 and RG 4.21.

8. The primary review organization, with the applicable secondary review organization, as necessary, verifies that isolation valves fail in the closed position to control the release of radioactive materials to the environment.

9. The primary review organization, the applicable secondary review organization, as necessary, and the organization responsible for the review of radiation protection verify that passive flow restrictions to limit reactor coolant loss from a rupture of the sample line are provided to keep radiation exposures to ALARA levels and the requirements of GDC 60 to control the release of radioactive materials to the environment.. The guidelines of Regulatory Position 2.i.(6) in RG 8.8 should be followed to meet this acceptance criterion. Redundant environmentally qualified, remotely operated isolation valves may replace passive flow restrictions in the sample lines to limit potential leakage. The automatic containment isolation valves should close on containment isolation signals or safety injection signals.

10. The primary review organization compares the seismic design and quality group classifications of the PSSs to the classifications of the fluid systems to which the sampling system is connected and confirms that the PSSs satisfy the acceptance criteria of Sections 3.2.1 and 3.2.2. The organization responsible for the review of instrumentation and control systems compares the seismic design and quality group classifications of the instruments for the PSSs to the classifications of the fluid systems to which the instruments of the sampling system is connected and confirms that the instrumentation satisfies the acceptance criteria of Section 3.10.

11. The organization responsible for the review of chemical engineering issues evaluates the Technical Specifications for process sampling to determine that their content and intent agree with the requirements developed as a result of the staff's review.

12. The primary review organization and the organizations responsible for the review of thermal-hydraulic performance of safety systems verify that provisions have been made to limit the potential for reactor coolant loss from the rupture of a sample line and provide the estimates of reactor coolant system fluid losses that would result from sample line rupture.

13. In lieu of the PASS, the organizations responsible for the review of emergency planning and for the review of chemical engineering issues should examine the following actions required to qualify process sampling for taking radioactive samples without having a specific post-accident sampling capability:

    A. Establish the capability for classifying a fuel damage event at the Alert level threshold

    B. Develop contingency plans for obtaining highly radioactive samples of the reactor coolant, containment sump, and containment atmosphere

    C. Determine for its own plant(s) that no decrease in the effectiveness of emergency plans will result from not having post-accident system capability

    D. Establish the capability to sample and analyze hydrogen in the containment atmosphere (recommended)

    E. Maintain offsite capability to monitor radioactivity, including radioactive iodines

14. The primary review organization and the organization responsible for the review of radiation protection verify that a leakage control program includes those portions of the

PSSs located outside of containment that contain or may contain radioactive material following an accident. The leakage control program should include periodic leak testing and measures to minimize leakage from the PSSs.

15. The organization responsible for the review of instrumentation and control systems reviews the instrumentation provided to ensure that information is provided for evaluating whether safety systems and other systems important to safety are performing their intended functions (i.e., reactivity control, fuel cladding integrity, maintaining reactor coolant system integrity, and maintaining containment integrity).

16. The organization responsible for the review of instrumentation and control systems reviews the sampling system related instrumentation to ensure that this instrumentation meets the guidance of RG 1.97.

17. For review of a DC application, the reviewer should follow the above procedures to verify the design, including requirements and restrictions (e.g., interface requirements and site parameters), set forth in the 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 shall 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 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 staff's technical review and analysis, as augmented by the application of programmatic requirements in accordance with the staff's review approach in the SRP 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 process sampling system includes piping, valves, heat exchangers, and other components from the point of sample withdrawal from a fluid system up to the analyzing station, sampling station, or local sampling point. Our review included the provisions proposed to sample all principal fluid process streams associated with plant operation and the applicant's proposed PSSs' designs. The review has included descriptive information for the process sampling system and the location of sampling points, as shown on schematics or P&IDs, if applicable. The basis for acceptance in our review has been conformance of the applicant's design for the process sampling system to applicable regulations, guides, and industry standards.

The staff concludes that the design of the process sampling system is acceptable and that the process sampling system meets the relevant requirements of 10 CFR 20.1101(b) and 10 CFR 20.1406 and GDCs 1, 2, 13, 14, 26, 41, 60, 63, and 64. The following paragraphs discuss the bases for this conclusion for each of the requirements, as applicable to iPWRs. For some designs, which may not contain the listed systems or locations for sampling, sampling of systems or locations found to be equivalent is acceptable in meeting the requirements of this DSRS section.

The staff has determined that the proposed process sampling system meets (1) the requirements of GDC 13 to monitor variables that can affect the fission process for normal operation, anticipated operational occurrences, and accident conditions by sampling the reactor coolant, the refueling water storage tank, the boric acid mix tank, and the boron injection tank for boron concentrations; (2) the requirements of GDC 13 and 14 to monitor variables that can affect the reactor coolant pressure boundary and to assure a low probability of abnormal leakage, rapidly propagating failure, and gross rupture, respectively, by sampling the reactor coolant and the secondary coolant for chemical impurities that can affect the reactor coolant pressure boundary; (3) the requirements of GDC 26 to control the rate of reactivity changes by sampling the reactor coolant, the emergency boration injection system boron injection tank, and the boric acid mix tank for boron concentration; (4) the requirements of GDC 13, and 14 to monitor variables that can affect the integrity of the reactor core and reactor coolant pressure boundary by sampling the chemical additive tank for chemical additive concentrations to ensure an adequate supply of chemicals for meeting material compatibility requirements; and (5) the requirements of GDC 64 to monitor for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents by sampling the reactor coolant, the pressurizer tank, the steam generator blowdown, the secondary coolant condensate treatment waste, and the containment atmosphere for radioactivity.

The staff has further determined that the proposed process sampling system meets (1) the requirements of 10 CFR 20.1101(b) to keep radiation exposures as low as is reasonably achievable, 10 CFR 20.1406 to minimize contamination of the systems with radioactive material, and of GDC 60 to control the release of radioactive materials to the environment by purging and draining sample streams back to the system of origin or to an appropriate radwaste treatment system, and by providing either redundant isolation valves that fail in the closed position or passive flow restrictions in the sampling lines; (2) the requirements of GDC 63 to detect conditions that may result in excessive radiation levels in fuel storage and radioactive waste systems by sampling the spent fuel pool water and the gaseous radwaste storage tank for radioactivity; and (3) the requirements of 10 CFR 50.34(f)(2)(xxvi) and related clarifications of Item III.D.1.1 in NUREG-0737 by inclusion of applicable portions of the systems in a leakage control program that contains periodic leak testing and measures to minimize the leakage from the systems.

The staff also has determined that the proposed process sampling system meets the quality standards requirements of GDC 1 and the seismic requirements of GDC 2 by designing the sampling lines and components of the process sampling system to conform to the classification of the system to which each sampling line and component is connected, in accordance with Regulatory Positions C.1, C.2, and C.3 of RG 1.26, Regulatory Positions C.1, C.2, C.3, and C.4 of RG 1.29, and the guidelines of RG 1.97.

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 Part 20, “Standards for Protection Against Radiation.”

2. 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities.”

3. 10 CFR Part 52, “Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants.”

4. GDC 61, “Fuel Storage and Handling and Radioactivity Control.”

5. GDC 19, “Control Room.”

6. GDC 4, “Environmental and Dynamic Effects Design Bases.”

7. RG 1.7, “Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident.”

8. RG 1.112, “Calculations of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors.”

9. RG 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors.”

10. ANSI/ANS Standard 18.1-1999, “Source Term Specification,” American National Standards Institute/American Nuclear Society.”

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

12. 40 CFR Part 190, “Environmental Radiation Protection Standards For Nuclear Power Operations.”

13. RG 1.89, “Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants.”

14. RG 1.143, “Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants.”

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

16. RG 1.29, “Seismic Design Classification.”

17. RG 1.117, “Tornado Design Classification.”

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

19. EPRI, “Pressurized Water Reactor Primary Water Chemistry Guidelines.”

20. EPRI, “Pressurized Water Reactor Primary Water Zinc Application Guidelines.”

21. EPRI, “Advanced Light Water Reactor Utility Requirements Document, Volume III, ALWR Passive Plant.”

22. NUREG-1242, “NRC Review of Electric Power Research Institute's Advanced Light Water Reactor Utility Requirements Document, Passive Plant Designs” Volume 3, Part 1 and Volume 3, Part 2 (ADAMS Accession Nos. ML070600372 and ML070600373).

23. EPRI, “Cobalt Reduction Guidelines.”

24. RG 8.8, “Information Relevant to Assuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as is Reasonably Achievable.”