**Draft for Comment**

U.S. NUCLEAR REGULATORY COMMISSION

# DESIGN-SPECIFIC REVIEW STANDARD FOR NuScale SMR DESIGN

## 9.3.4 CHEMICAL AND VOLUME CONTROL SYSTEM

**REVIEW RESPONSIBILITIES**

**Primary -** Organization responsible for the review of PWR reactor coolant systems

**Secondary -** Organization responsible for reactor coolant chemistry

### I. AREAS OF REVIEW

The NuScale small modular reactor (SMR) plants include a chemical and volume control system (CVCS). This system maintains the required water inventory and quality in the reactor coolant system (RCS), provides water flow to the pressurizer spray, controls the boric acid concentration in the reactor coolant, adds heat and establishes stable natural circulation flow during reactor startup, controls the primary water chemistry, and reduces coolant radioactivity levels. In the NuScale design, the CVCS does not perform or support the safety functions of emergency boration or core cooling as part of emergency core cooling system (ECCS) for mitigation of the design basis events.

The staff reviews the system from the letdown line of the primary system to the charging lines that provide makeup to the primary system. The system is reviewed to the interfaces with the demineralized water makeup system and radioactive waste system. The review is performed to assure conformance with the requirements of General Design Criteria (GDC) 1, 2, 5, 14, 29, 33, 60, and 61.

The CVCS has been categorized as a non-safety system; however, there are CVCS components directly connected to the reactor coolant pressure boundary that provide reactor coolant pressure boundary protection and isolation functions are safety-related and risk-significant. The system is risk-significant as it provides makeup water to the RCS ensuring adequate coolant inventory against small breaks and leakage from the RCPB and to reduce the radioactivity level of the primary coolant system.

The specific areas of review are as follows:

1. The safety-related functional performance characteristics of CVCS components and the effects of adverse environmental occurrences, abnormal operational requirements, or accident conditions such as those due to a loss-of-coolant accident (LOCA).

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2. The determination that a malfunction or single failure of an active component or the loss of a cooling source will not reduce the safety-related functional performance capabilities of the system.

3. That quality group and seismic design requirements are met and the effects of failure of equipment or components not designed to withstand seismic events on safety-related functions of the system are evaluated.

4. The system features provided to prevent precipitation of boric acid in components and lines containing boric acid solutions, and assess the adequacy of the system design to allow personnel access considering the effects of toxic, irritating, or explosive chemicals that may be used.

5. Provisions for operational testing and the instrumentation and control features that determine and verify that the system is operating in the correct mode.

6. Provisions to prevent the formation of such vacuum conditions that could cause wall inward buckling and failure in tanks that can contain primary system water.

7. <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 SSCs related to this DSRS section in accordance with DSRS 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 DSRS 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.

<u>Review Interfaces:</u>

Other DSRS sections interface with this section as follows:

1. Determination of the acceptability of the seismic and quality group classifications for systems and components is performed under DSRS Sections 3.2.1 and 3.2.2.

2. Determination of the acceptability of the design analysis, procedures, and criteria used to establish the ability of seismic Category I structures housing the system and supporting systems to withstand the effects of natural phenomena, such as the safe shutdown earthquake (SSE), the probable maximum flood (PMF), and tornado missiles

is performed under DSRS Sections 3.3.1, 3.3.2, 3.4.2, 3.5.3, 3.7.1, 3.7.2, 3.7.3, 3.8.4, and 3.8.5.

3. Evaluation of the capability of the CVCS to withstand external and internal flood conditions is performed under DSRS Sections 3.4.1 and 9.3.3.

4. Evaluation of the capability of structures and barriers designed to protect the CVCS from internally generated missiles both inside and outside primary containment is performed under DSRS Sections 3.5.1.1 and 3.5.1.2.

5. Evaluation of the capability of safety-related systems to withstand the effects of missiles generated by natural phenomena or externally generated missiles is performed under DSRS Sections 3.5.1.4 and 3.5.2.

6. Evaluation of the effect of high- and moderate-energy CVCS system piping failures outside containment to assure that other safety-related systems will not be made inoperable is performed under DSRS Section 3.6.1.

7. Determination that the piping, components, and structures are designed in accordance with applicable codes and standards are performed under DSRS Sections 3.9.1, 3.9.2, 3.9.3, 5.2.2, and 5.2.3

8. Review of the adequacy of the inservice testing program of pumps and valves is performed under DSRS Section 3.9.6.

9. Review of the seismic qualification of Category I instrumentation and electric equipment is performed by under DSRS Sections 3.10.

10. Review of the environmental qualification of mechanical and electrical safety-related equipment is performed under DSRS Section 3.11.

11. Reviews of liquid, solid, and gaseous waste treatment aspects of the CVCS are performed under DSRS Sections 11.2, 11.3, and 11.4, respectively.

12. Reviews of the CVCS to verify that low-pressure portions of the CVCS that interface with the RCS are designed, to the extent practical, to withstand full RCS pressure. If designing the CVCS with an ultimate rupture strength capable of withstanding full RCS pressure is not possible, the reviewer verifies that appropriate compensating measures have been taken in accordance with the review provided in DSRS Section 3.12.

13. Evaluation of the injection of borated water into the RCS to meet combined reactivity control system redundancy and capability requirements of GDC 26 and 27 is performed under DSRS Section 4.3.

14. Review of the CVCS to verify that low temperature overpressure protection requirements are met is performed under DSRS Section 5.2.2.

15. For the plant designs that rely on the CVCS to perform or support safety functions of emergency boration and core cooling as part of ECCS for mitigation of the design basis events, review of the CVCS flow capacity and injection pressure to verify that specified

acceptable fuel design limits are not exceeded following an inadvertent opening of a pressurizer relief valve, and that the acceptance criteria of 10 CFR 50.46 are complied with following a postulated LOCA in evaluating the ECCS function is performed under DSRS Sections 6.3, 15.6.1, and 15.6.5.

16. Verification that inservice nondestructive examination requirements are met for system components is performed under DSRS Sections 5.2.4 and 6.6.

17. Review of the design of isolation provisions of those portions of the CVCS that penetrate primary containment is performed under DSRS Section 6.2.4.

18. Evaluation of the adequacy of the design, installation, inspection, and testing of all instrumentation, sensing, and controls required to provide the safety-related and risk-significant functions of the CVCS is performed under DSRS Sections 7.1, 7.6, and Appendix 7A.

19. Evaluation of the adequacy of the design, installation, inspection, and testing of all electrical systems required to provide the safety-related and risk-significant functions of the CVCS is performed under Sections 8.3.1 and 8.3.2.

20. Review of the capability of the component cooling water system essential for CVCS operation is performed under DSRS Section 9.2.2.

21. Review of the CVCS with respect to fire protection is performed under DSRS Section 9.5.1.

22. Reviews of liquid, solid, and gaseous waste treatment aspects of the CVCS are performed under DSRS Sections 11.2, 11.3, and 11.4, respectively.

23. Review of the process and effluent radiological monitoring aspects of the CVCS is performed under DSRS Section 11.5.

24. Review of the system with respect to maintaining occupational radiation exposure as low as reasonably achievable (ALARA) and to providing radiation protection design features, respectively, is performed under DSRS Sections 12.1 and 12.3.

25. Review of CVCS malfunctions that can result in a decrease in boron concentration in the reactor coolant to assure that fuel damage limits are not exceeded and that adequate time is available to terminate the dilution before the shutdown margin has been eliminated is performed under DSRS Section 15.4.6.

26. Review of DSRS technical specifications is coordinated and performed under DSRS Section 16.0.

27. Review of the reliability assurance program is coordinated and performed under SRP Section 17.4.

28. Review of quality assurance is coordinated and performed under SRP Section 17.5.

29. Review of regulatory treatment of nonsafety systems (RTNSS) is coordinated and performed under SRP Section 19.3.

## 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 5, Sharing of Structures, Systems, and Components.

5. GDC 14, Reactor Coolant Pressure Boundary.

6. GDC 29, Protection Against Anticipated Operational Occurrences.

7. GDCs 33, Reactor Coolant Makeup.

8. GDCs 60 Control of Releases of Radioactive Materials to the Environment.

9. GDC 61, Fuel Storage and Handling and Radioactivity Control.

10. 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.

11. 10 CFR 50.34(f)(2)(xxvi), with respect to the provisions for a leakage detection and control program to minimize the leakage from those portions of the CVCS outside of the containment that contain or may contain radioactive material following an accident.

12. 10 CFR 52.47(b)(1), which requires that a DC application contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.

13. 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.

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 CVCS safety-related functional performance should be maintained in the event of adverse environmental phenomena such as earthquakes, tornadoes, hurricanes, and floods, or in the event of certain pipe breaks or loss of offsite power. For compliance with GDC 29, and 33 the CVCS should provide sufficient pumping capacity to supply borated water to the RCS, maintain RCS water inventory within the allowable pressurizer level range for all normal modes of operation, and supply reactor coolant makeup as a defense in depth function in the event of small leakage from the RCS.

2. SECY-77-439 describes the concept of single failure criteria and the application of the single failure criterion that involves a systematic search for potential single failure points and their effects on prescribed missions. Application of the single failure assumption in system design and analysis provides redundancy and defense-in-depth to ensure functional performance of the CVCS.

   Also, the requirements of GDC 5 prohibiting the sharing among nuclear units the SSCs important to safety would be met by the use of a separate CVCS for each unit.

3. 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 in leakage 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.

4. 10 CFR 50.55(a) requires that components of the RCPB be designed, fabricated, erected, and tested in accordance with the requirements for Class 1 components of Section III of the ASME Boiler and Pressure Vessel Code or equivalent quality standards. Regulatory Guide 1.26 describes a quality classification system that may be used to determine quality standards acceptable to the NRC staff for satisfying GDC 1 for other safety related components containing water, steam, or radioactive materials in light-water-cooled nuclear power plants. RG 1.29 describes a method acceptable to the NRC staff for identifying and classifying those features of LWRs that should be designed to withstand the effects of the safe shutdown earthquake (SSE).

   The requirements of GDC 1 regarding the quality standard are met by acceptable application of quality group classifications and application of quality standards as described in RG 1.26. The requirement of GDC 2 regarding the protection against natural phenomena are met by meeting the guidance of RG 1.29, Position C.1, for safety-related portions of the system and Position C.2 for nonsafety-related portions.

5. The CVCS design and arrangement should be that all components and piping that can contain boric acid will either be heat traced, located within heated rooms, or maintained at a low enough concentration to prevent precipitation of boric acid.

   As additional specific criteria used to review the CVCS design, the CVCS should include provisions for monitoring: (a) temperature upstream of the demineralizer to assure that resin temperature limits are not exceeded, and (b) filter demineralizer differential pressure to assure that pressure differential limits are not exceeded. In addition, the CVCS should have provision for automatically diverting or isolating the CVCS flow to the demineralizer in the event the demineralizer influent temperature exceeds the resin temperature limit.

6. 10 CFR 50.34(f)(2)(xxvi), as applicable, specifies the provisions regarding detection of reactor coolant leakage outside containment. These requirements will be met, in part, by providing leakage control and detection systems in the CVCS and implementation of appropriate leakage control program.

7. Implementation of Action 1 specified in Bulletin 80-05 provides an acceptable means for the system to prevent the CVCS holdup tanks, which can contain radioactive release, from the formation of such vacuum conditions that could cause wall inward buckling and failure.

   The requirements of GDC 60 and 61 can be met, in part, by providing in the CVCS appropriately designed venting and draining closed systems to confine the radioactivity associated with the effluents.

8. 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 constructed and will operate in accordance with the design certification. 10 CFR 52.97(b)(1) 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 14.3 provides guidance for reviewing the ITAAC. The requirements of 10 CFR 52.47(a)(1)(vi) and 10 CFR 52.97(b)(1) will be met, in part, by identifying ITAAC of the top-level design features of the CVCS 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. GDC 1 requires that structures, systems, and components (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 CVCS may be important to safety in that: 1) the CVCS may provide a means of makeup for the RCS coolant inventory in the event of small leaks; 2) the CVCS may be capable of borating the RCS to a safe cold shutdown condition; 3) the CVCS is relied upon to control RCS water chemistry to maintain the integrity of the RCS pressure boundary; 4) through connections to the RCS a CVCS failure could adversely affect the integrity of the RCS or

containment systems and 5) portions of the CVCS may contain radioactive material. Meeting the requirements of GDC 1 (and the guidance of RG 1.26) ensures that the CVCS will be designed, fabricated, erected and tested to generally accepted and recognized codes and standards that are sufficient to assure a quality system in keeping with the importance of the designated safety functions.

2. GDC 2 requires that SSCs important to safety be designed to withstand the effects of natural phenomena without the loss of capability to perform their safety functions. Certain portions of the CVCS may have functions important to plant safety that should be designed to withstand the safe shutdown earthquake (SSE). RG 1.29 provides guidance for determining which systems should be designated Seismic Category I; position C.1 provides guidance for safety related portions and position C.2 provides guidance for nonsafety related portions. For example, the CVCS connects to the RCS, and components that form interfaces between Seismic Category I and non-Seismic Category I features should be designed to Seismic Category I requirements. Meeting the requirements of GDC 2 (and the guidance of RG 1.29) will enhance plant safety by ensuring the integrity of Seismic Category I portions of the system during a design basis seismic event.

3. GDC 5 prohibits the sharing of SSCs among nuclear power units unless it can be shown that such sharing will not significantly impair their 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. The CVCS may be designed to provide essential safety-related or risk significant functions necessary for continued safe operation of the unit(s), such as protection and isolation of the reactor coolant pressure boundary and control of in-system radioactivity. The CVCS must be designed such that the ability to perform these and other designated safety-related or risk significant functions is not compromised for each unit regardless of equipment failures or other events that may occur in another unit. Meeting the requirements of GDC 5 provides assurance that unacceptable effects of equipment failures or other events occurring in one unit of a multi-unit site will not propagate to the unaffected unit(s).

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

5. GDC 29 requires that the reactivity control systems be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences (AOO). Portions of the CVCS may be relied upon to provide or support negative reactivity addition by injection of boric acid to the RCS, and assures that specified acceptable fuel design limits (SAFDLs) will not be exceeded. Meeting the requirements of GDC 29 enhances plant safety by assuring that the reactivity control aspects of the CVCS will have a high probability of injecting sufficient negative reactivity to prevent exceeding SAFDLs during AOOs, thereby preventing damage to the fuel matrix and cladding.

6. GDC 33 requires that a system be provided to supply reactor coolant makeup for theprotection against small breaks in the RCPB. The CVCS may be relied upon to provide charging and makeup in the event of small leakage from the RCPB and rupture of small piping or components that are part of the pressure boundary. Meeting the requirements of GDC 33 enhances plant safety by ensuring that the CVCS can provide sufficient makeup capacity to maintain the required RCS water inventory and prevent the violation of SAFDLs given a small break in the reactor coolant pressure.

8. GDC 60 requires that the release of radioactive material to the environment be controlled. The CVCS during normal reactor operation can contain radioactive material in gaseous and liquid forms. The CVCS is designed with storage tanks to handle venting and draining from various CVCS systems. The CVCS vent and drain systems are designed to appropriately confine the radioactivity associated with the effluents. Meeting the requirements of GDC 60 enhances plant safety by preventing the uncontrolled release of radioactive material to the environment.

9. GDC 61 requires that systems that may contain radioactivity be designed to assure adequate safety under normal and postulated accident conditions. The CVCS is connected to the RCS and during normal and postulated accident conditions may contain radioactivity throughout the system. Meeting the requirements of GDC 61 ensures that applicable portions of the CVCS are designed to provide confinement of radioactive material and to reduce the potential exposure to radioactive materials to the lowest practical levels.

10. 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 Primary Shutdown Systems (PSS) 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.

## 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.

For the purpose of this DSRS section, it is assumed that the CVCS consists of: regenerative and non-regenerative heat exchangers to cool the letdown flow from the RCS before processing

through the demineralizers (or ion exchangers) and to reheat it prior to reinjection into the RCS; demineralizers and filters for removal of suspended and dissolved impurities to reduce -coolant radioactivity levels; pumps which inject makeup flow into the RCS; and the volume control tank for system surge capacity and makeup volume. Other CVCS functions, such as pressurizer spray, also make use of CVCS components.

The CVCS varies from the traditional PWR chemical volume and control system in that the NuScale reactor design uses natural circulation core flow instead of reactor coolant pumps; therefore, there is no need for the traditional CVCS reactor coolant pump seal water injection function. The CVCS does provide boron and RCS level control during normal operation. The CVCS does not perform safety-related accident mitigation functions (e.g., coolant injection) for design basis events.

The system design should meet the acceptance criteria given 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.

1. In accordance with 10 CFR 52.47(a)(8),(21), and (22), and 10 CFR 52.79(a)(17), (20) and (37) for new reactor 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 that are identified in the version of NUREG-0933 current on the date up to six months before application and that 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). 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.

2. The staff reviews the safety analysis report (SAR) to determine that the system description and schematics or piping and instrumentation diagrams (P&IDs), if applicable, show the CVCS equipment that is used for normal operation and the minimum system heat transfer and flow requirements for normal operation. The system performance specifications as stated in the DC will also be reviewed to determine that it limits expected component operational degradation (e.g., pump leakage, heat exchanger scaling, resin deterioration) and describes the procedures that will be followed to detect and correct these conditions when they become excessive.

   The system description and schematics or P&IDs, if applicable, layout drawings, and component descriptions and characteristics are reviewed for the following points:

   A. Essential portions of the CVCS that provide reactor coolant boundary protection functions are correctly identified and are verified to be isolable from the nonessential portions of the system and from interfacing systems such as demineralized water makeup and radioactive waste systems. The documentation will be reviewed to verify that they clearly indicate physical divisions between such portions and indicate design classification changes. Systems drawings are also reviewed to see whether they show the means for accomplishing isolation and the system description is reviewed to identify minimum performance specifications for the isolation valves.

   B. The review is performed to assure that essential portions of the CVCS that provide reactor coolant boundary protection functions, including the isolation valves separating essential portions from nonessential portions are classified Quality Group A, B, or C and seismic Category I in accordance with the guidelines of Regulatory Guides 1.26 and 1.29; also, system descriptions in the SAR are reviewed to verify that the above seismic and safety classifications have been included, and that the system description and schematics or P&IDs, if applicable, indicate any points of change in piping quality group classification.

   C. The failure of portions of the system or of other systems not designed to seismic Category I standards and located close to essential portions of the system, or of nonseismic Category I structures that house, support, or are close to essential

portions of the CVCS, will not preclude operation of the essential portions of the CVCS (Position C.2 of Regulatory Guide 1.29). Reference to SAR sections describing site features and the general arrangement and layout drawings should be provided, as well as the SAR tabulation of seismic design classifications for structures and systems. Statements in the SAR that verify that the above conditions are met are acceptable.

D. The reviewer verifies the adequacy of the system for reactivity control in the following areas:

  i. Boration of the reactor coolant system is accomplished with boric acid source, such as boric acid storage tank (BAST) or refueling water storage tank (RWST), where applicable, through redundant flow paths, and CVCS meets PWR boration technical specifications. This is verified from the review of the system description and schematics or P&IDs, if applicable, and system description.

  ii. The amount and concentration of boric acid stored in the CVCS, such as BAST or RWST exceeds the amount required to borate the reactor coolant system to cold shutdown conditions assuming that the control rod assembly with the highest reactivity worth is held in the fully withdrawn position, and to compensate for subsequent xenon decay during any part of core life.

E. The adequacy of the CVCS for control of water chemistry is verified by examination of the information provided in the SAR (i.e., the allowable ranges for primary coolant activity, total dissolved solids, pH, and maximum allowable oxygen and halide concentrations and verification that the CVCS can meet industry guidelines and reactor plant supplier recommendations for PWR reactor coolant system water chemistry). The reviewer verifies that the primary water chemistry program provides for compatibility with materials to be exposed to reactor coolant under the expected service conditions. The reviewer evaluates the proposed chemistry program with respect to that described in the latest version in the Electric Power Research Institute (EPRI) report series, "PWR Primary Water Guidelines" (Ref. 25).

F. The adequacy of resin over-temperature protection is verified by reviewing the system description and drawings to determine that temperature sensors are provided that will actuate the demineralizer bypass or isolation valves. Also, verify that instrumentation is available to monitor filter and demineralizer differential pressures. Acceptability may be determined by identification of programmatic controls that adequately tests temperature and pressure instrumentation actuations to bypass or isolate demineralizers on high temperature or differential pressure.

G. The system description and schematics or P&IDs, if applicable, are examined to determine whether all components and piping that can contain boric acid will either be heat traced or will be located within heated rooms to prevent precipitation of boric acid or maintained at a low enough concentration to preclude the need for heat tracing or heated rooms.

H. The application is reviewed with respect to establishing a leakage control program. Inside containment leakage control programs shall be addressed in plant technical specifications. For those portions of the CVCS located outside containment that may contain radioactive material following an accident, a leakage control program shall be established by plant operational programs consistent with item III.D.1.1 of NUREG-0737.

I. The CVCS low pressure or holdup tanks that can contain primary system water are reviewed to assure adequate measures have been taken to protect against vacuum conditions that could result in tank damage (see Reference 21). With respect to the prevention of vacuum conditions in system tanks, the reviewer should consider the following: (a) tanks with a cover gas are able to admit the cover gas fast enough to keep up with the maximum rate of liquid removal; (b) vacuum relief valves are included in a surveillance program; and (c) tanks subject to freezing conditions have adequate freeze protection for the tank and the vacuum relief system.

4. The reviewer verifies that the safety-related functions of the system will be maintained as required in the event of adverse environmental phenomena such as earthquakes, tornadoes, hurricanes, and floods, or in the event of certain pipe breaks or loss of offsite power. The reviewer uses engineering judgment, failure modes and effects analyses, and the results of reviews performed under other DSRS sections as applicable to determine the following:

A. The system description and drawings are reviewed in conjunction with the reactor coolant system to determine whether the CVCS has sufficient pumping capacity to maintain the RCS water inventory within the allowable pressurizer level range for all normal modes of operation, including startup from cold shutdown, full power operation, and plant cooldown.

B. Essential safety-related components of the CVCS can function as required in the event of loss of offsite power. The SAR is reviewed to verify that for each CVCS component or subsystem affected by the loss of offsite power, boric acid addition and coolant charging capabilities meet or exceed minimum specifications as stated in the DC. Statements in the SAR and the results of failure modes and effect analyses are considered in assuring that the system meets these specifications. This will be acceptable verification of system functional reliability.

5. The descriptive information, schematics or P&IDs (if applicable), layout drawings, and failure modes and effects analyses in the SAR are reviewed to assure that safety-related portions of the system (reactor coolant pressure boundary and isolation of the primary coolant system) will function following design basis accidents assuming a single active component failure. The reviewer evaluates the analyses presented in the SAR to assure function of required components, traces the availability of these components on system drawings, and checks that the SAR contains verification that minimum system requirements are met for each accident situation for the required time spans. For each case, the design will be acceptable if minimum system specifications are met.

6. The boron recovery system (BRS), if applicable, is not required for safe shutdown, or for the prevention or mitigation of postulated accidents. The BRS will be reviewed for the following: if the system tankage is of nonseismic Category I design, the results of analyses which postulate the rupture of tanks are reviewed to verify that the accident releases are in accordance with safe limits.

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

8. 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 final safety analysis report (FSAR) meets the acceptance criteria. DCs have referred to the FSAR as 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 should be added to the DC FSAR.

   For review of a COL application, the scope of the review is dependent on whether the COL applicant references a DC, an early site permit (ESP) or other NRC approvals (e.g., manufacturing license, site suitability report or topical report).

9. 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 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 chemical and volume control system (including boron recovery system, if applicable) includes components and piping associated with the system from the letdown line of the primary system to the charging lines that provide makeup to the primary system. Based on the review of the applicant's proposed design criteria, design bases, safety classification for the chemical and volume control system, the requirements for system performance of necessary functions during normal, abnormal, and accident conditions, and the applicable programmatic requirements the staff concludes that the design of the chemical and volume control system and supporting system is acceptable and meets the requirements of General Design Criteria 1, 2, 5, 14, 29, 33, 60, and 61, and 10 CFR 50.34(f)(2)(xxvi) and 10 CFR 20.1406.

This conclusion is based on the following: the applicant's design of the chemical and volume control system meets (1) the requirements of General Design Criterion 1 and the guidelines of Regulatory Guide 1.26 by assigning quality group classifications to system components in accordance with the importance of the safety function to be performed; (2) the requirements of General Design Criterion 2 and the guidelines of Regulatory

Guide 1.29 by designing safety-related portions of the system to seismic Category I requirements; (3) the requirements of General Design Criterion 5 by designing the CVCS so that components important to safety are not shared between nuclear power units unless such sharing will not significantly impair the ability of the CVCS to perform its safety functions in the event of an accident in one unit and an orderly shutdown and cooldown of the remaining units; (4) the requirements of General Design Criterion 14 by maintaining reactor coolant purity and material compatibility to reduce corrosion and thus reduce the probability of abnormal leakage, rapid propagating failure, or gross rupture of the reactor coolant pressure boundary; (5) the requirements of General Design Criterion 29 as related to the reliability of the CVCS to provide negative reactivity to the reactor by supplying borated water to the reactor coolant system in the event of anticipated operational occurrences; (6) the requirements of General Design Criteria 33 by designing the CVCS with the capability to supply reactor coolant makeup in the event of small breaks or leaks in the reactor coolant pressure boundary (7) the requirements of General Design Criteria 60 and 61 with respect to confining radioactivity by venting and collecting drainage from the CVCS components through closed systems; (8) the requirements of 10 CFR 20.1406 to minimize contamination of systems with radioactive material, and (10) 10 CFR 50.34(f)(2)(xxvi) for applicants subject to 10 CFR 50.34(f), with respect to leakage detection and control in the design of CVCS systems outside containment that contain (or may contain) radioactive material following an accident.

For DC and COL reviews, the findings will also summarize the staff's evaluation of requirements and restriction (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.”