Draft for Comment

U.S. NUCLEAR REGULATORY COMMISSION

DESIGN-SPECIFIC REVIEW STANDARD FOR NuScale SMR DESIGN

12.2 RADIATION SOURCES

REVIEW RESPONSIBILITIES

Primary - Organization responsible for the review of health physics issues.

Secondary - None

I. AREAS OF REVIEW

The staff will review the applicant's final safety analysis report (FSAR) for a design certification (DC) or combined license (COL), as it relates to radiation sources in normal operations, anticipated operational occurrences (AOOs), and accident conditions used as the bases for determining the radiation protection design features provided to ensure compliance with the requirements of the Commission, including: Title 10 of the *Code of Federal Regulations* (CFR), Section 20.1101(b) to maintain Occupational Radiation Exposure (ORE) as low as is reasonably achievable (ALARA); 10 CFR Part 20, Subpart H to control the concentration of radioactive material in the air for workers; 10 CFR 20.1301(e) and 40 CFR Part 190 to control the exposure of members of the public from direct radiation sources. The staff also reviews sources of radioactive material used as the bases for determining the design features needed to comply with the requirements of: 10 CFR 20.1406 to minimize contamination of the facility and the environment, minimize the generation of waste and facilitate decommissioning; 10 CFR Part 20, Subpart I to control radioactive material, and 10 CFR 50.49 to identify the types and quantities of radioactive material that may affect the qualification of electrical equipment important to safety.

The specific areas of review are as follows:

1. Contained Sources. The description of radiation sources, during normal operations and accident conditions in the plant, is used as the basis for designing the radiation protection program and for shield design calculations. FSAR Chapter 11 describes the sources contained in equipment of the radioactive waste management systems. This description should include isotopic composition, location in the plant, source strength and source geometry, and the basis for the values (in the DC FSAR or COL FSAR or, for sources not described in a referenced certified design, the COL FSAR). The descriptions should include any required radiation sources containing byproduct, source, and special nuclear materials.

2. Airborne Radioactive Material Sources. The staff will review the description of airborne radioactive material sources in the plant considered in the design of the ventilation systems and used for the design of personnel protective measures and for dose assessment. (FSAR Chapter 11 contains the description for airborne sources to be considered for their contribution to the plant effluent releases, through equipment of the

radioactive waste management systems or the plant ventilation system.) This description should include a tabulation of the calculated concentrations of radioactive material, by nuclide, expected during normal operation, AOOs, and accident conditions for equipment cubicles, corridors, and operating areas normally occupied by operating personnel. It should also include models and parameters for the calculations (DC FSAR or COL FSAR or, for sources not described in a referenced certified design, the COL FSAR).

3. Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC). For DC and COL reviews, the staff reviews the applicant's proposed ITAAC associated with the structures, systems, and components (SSCs) related to this design-specific review standard (DSRS) section in accordance with Standard Review Plan (SRP) Section 14.3, "Inspections, Tests, Analyses, and Acceptance Criteria." The staff recognizes that the review of ITAAC cannot be completed until after the rest of this portion of the application has been reviewed against acceptance criteria contained in this DSRS section. Furthermore, the staff reviews the ITAAC to ensure that all SSCs in this area of review are identified and addressed as appropriate in accordance with SRP Section 14.3.

4. COL Action Items and Certification Requirements and Restrictions. For a DC application, the review will also address COL action items and requirements and restrictions (e.g., interface requirements and site parameters).

   For a COL application referencing a DC, a COL applicant must address COL action items (referred to as COL license information in certain DCs) included in the referenced DC. Additionally, a COL applicant must address requirements and restrictions (e.g., interface requirements and site parameters) included in the referenced DC.

Review Interfaces

Systems described in the FSAR may differ from those outlined in the DSRS. The staff should use the following recommended section interfaces as the basis for reviewing other supplemental or complementary information provided in the FSAR for the specific plant design to identify and quantify sources of radioactivity that may require radiation protection design features to minimize ORE and protect equipment:

1. 3.8.2 STEEL CONTAINMENT – as it relates to the types and quantities of radioactive materials, and the associated bases, resulting from the activation of the materials selected for the steel containment vessel, including the contribution of corrosion and activation products to the fuel pool radioactive material inventory.

2. 3.2.1 SYSTEM QUALITY GROUP CLASSIFICATION – as it relates to the seismic classification of SSCs containing radioactive material.

3. 3.2.2 SYSTEM QUALITY GROUP CLASSIFICATION – as it relates to the quality group classification of SSCs containing radioactive material.

4. 3.11 ENVIRONMENTAL QUALIFICATION OF MECHANICAL AND ELECTRICAL EQUIPMENT – as it relates to sources of radiation that are used to establish the design basis for the radiation dose to equipment.

5. 4.2 FUEL SYSTEM DESIGN – as it relates to the bases for determining the radioactive

content of: irradiated fuel, including fuel burn up, fuel enrichment, fuel bundle materials; irradiated control rods materials; and fuel power density as it relates to the potential amount of radioactive material created by the activation of material deposited in high neutron flux areas, which is available for release into the Reactor Coolant System (RCS) fluid.

6. 4.3 NUCLEAR DESIGN – as it relates to the bases for determining the radioactive content of irradiated core components, such as: mobile or fixed neutron detectors; primary or secondary neutron startup sources; instrumentation internal to the reactor vessel, created by the activation of material located in high neutron flux areas, which is available for release into the Reactor Coolant System (RCS) fluid, or represents a occupational radiation protection hazard.

7. 4.5.1 CONTROL ROD DRIVE STRUCTURAL MATERIALS – as it relates to the types and quantities of radioactive materials, and the associated bases, resulting from the activation of the materials selected for the control rod drive mechanisms up to the coupling interface with the reactivity control (poison) elements in the reactor vessel.

8. 4.5.2. REACTOR INTERNAL AND CORE SUPPORT STRUCTURE MATERIALS – as it relates to the introduction rates (e.g., cobalt content) of material contributing to ORE and the quantity and isotopic content of irradiated start up neutron sources and neutron detection equipment (e.g., in core neutron detectors).

9. 5.2.3 REACTOR COOLANT PRESSURE BOUNDARY MATERIALS – as it relates to the introduction rates of material (e.g., cobalt content) contributing to ORE from RCS pressure boundary components.

10. 5.3.1 REACTOR VESSEL MATERIALS – as it relates to the introduction rates of material (e.g., cobalt content) contributing to ORE from reactor pressure vessel materials in contact with RCS fluids.

11. 5.4 REACTOR COOLANT SYSTEM COMPONENT AND SUBSYSTEM DESIGN – as it relates to the types, quantities and associated bases for radioactive fission, activation and corrosion products contained in the tanks, heat exchangers and related components which interface with the RCS.

12. 5.4.7 RESIDUAL HEAT REMOVAL (RHR) SYSTEM – as it relates to the types, quantities and associated bases for radioactive fission, activation and corrosion products contained in the piping, heat exchangers, pumps and other components of the system.

13. 6.1.1 ENGINEERED SAFETY FEATURES MATERIALS – as it relates to the types, quantities and associated bases for radioactive fission, activation and corrosion products contained in the piping, heat exchangers and other components of the system.

14. 6.2.8 CONTAINMENT EVACUATION SYSTEM – as it relates to; the types, quantities and associated bases for radioactive fission, activation and corrosion products contained in the piping, heat exchangers and other components of the system, that contribute to ORE; to the extent that the instrumentation credited for RCS leakage detection is described in FSAR Chapter 12, the types, quantities and associated bases for radioactive materials used to determine the required sensitivity of radiation monitoring equipment.

15. 6.3 EMERGENCY CORE COOLING SYSTEM – as it relates to the types, quantities and associated bases for radioactive fission, activation and corrosion products contained in the piping, heat exchangers and other components of the system.

16. 9.1.2 NEW AND SPENT FUEL STORAGE – as it relates to determining ORE from; permanent or temporary irradiated fuel storage and handling locations; the quantity, isotopic content, enrichment and burn up of irradiated fuel assemblies; the location and isotopic content, and the associated bases, of irradiated components.

17. 9.1.3 SPENT FUEL POOL COOLING AND CLEANUP SYSTEM – as it relates to determining the bases (i.e. enrichment, burn up and isotopic content and decontamination factors) for ORE from; permanent or temporary irradiated fuel storage and handling locations; irradiated components storage; concentrations of radionuclides contained within the refueling and fuel storage pools; the amount of radioactive fission, activation and corrosion products contained in filtration media and purification media; and radioactive material resulting from direct neutron activation of the contents of the reactor module pool fluid.

18. 9.2.2 REACTOR AUXILIARY COOLING WATER SYSTEM – as it relates to the types, quantities and associated bases for radioactive fission, activation and corrosion products contained in the system piping, tanks, vessels, filtration media and purification media.

19. 9.2.6 CONDENSATE STORAGE FACILITIES – as it relates to the types, quantities and associated bases for radioactive fission, activation and corrosion products, including those from direct neutron activation of the contents of the secondary fluid systems located within the reactor or containment vessels, contained in the system piping, tanks, vessels, filtration media and purification media.

20. 9.3.2 PROCESS AND POST-ACCIDENT SAMPLING SYSTEMS – as it relates to the types and quantities of radioactive fission, activation and corrosion products that may accumulate in the system piping, tanks and vessels as a result of purging or sampling.

21. 9.3.4 CHEMICAL AND VOLUME CONTROL SYSTEM (PWR) (INCLUDING BORON RECOVERY SYSTEM) – as it relates to the types and quantities of radioactive fission, activation and corrosion products within the system piping tanks and vessels, filtration media and purification media.

22. 9.4.1 CONTROL ROOM AREA VENTILATION SYSTEM – as it relates to the types and quantities of radioactive fission, activation and corrosion products within the ventilation system filtration media and associated structures.

23. 9.4.2 SPENT FUEL POOL AREA VENTILATION SYSTEM – as it relates to the types and quantities of radioactive fission, activation and corrosion products within the ventilation filtration media and associated structures.

24. 9.4.3 AUXILIARY AND RADWASTE AREA VENTILATION SYSTEM – as it relates to the types and quantities of radioactive fission, activation and corrosion products within the ventilation filtration media and associated structures.

25. 10.2 TURBINE GENERATOR – as it relates to the types, quantities and concentrations

of radioactive fission, activation and corrosion products within the system piping resulting from direct neutron activation of the contents of the secondary fluid systems contained within the reactor vessel.

26. 10.3 MAIN STEAM SUPPLY SYSTEM – as it relates to the types, quantities and concentrations of radioactive fission, activation and corrosion products within the system piping resulting from direct neutron activation of the contents of the secondary fluid systems contained within the reactor vessel.

27. 10.4.1 MAIN CONDENSERS – as it relates to the types, quantities and concentrations of radioactive fission, activation and corrosion products within the system resulting from direct neutron activation of the contents of the secondary fluid systems contained within the reactor vessel.

28. 10.4.6 CONDENSATE CLEANUP SYSTEM – as it relates to the types, quantities and concentrations of radioactive fission, activation and corrosion products within the system piping tanks and vessels, filtration media and purification media, as determined from the bases, models and assumptions described in Chapter 11, and activation products resulting from direct neutron activation of the contents of the secondary fluid systems.

29. 10.4.8 STEAM GENERATOR BLOWDOWN SYSTEM – as it relates to the types, quantities and concentrations of radioactive fission, activation and corrosion products within the system piping tanks and vessels, filtration media and purification media, as determined from the bases, models and assumptions described in Chapter 11.

30. 11 RADIOACTIVE WASTE MANAGEMENT – as it relates to the description of the methods, models and assumptions used as the bases for determining concentrations and quantities of radioactive material in SSCs described in Section 12.2 of the FSAR, also, as it relates to the description of sources contained in equipment of the radioactive waste management system.

31. 16.0 TECHNICAL SPECIFICATIONS – as it relates to the types, quantities and concentrations of radioactive fission, activation and corrosion products within the system piping tanks and vessels, filtration media and purification media, that may be derived from allowable primary-to-secondary leakage or secondary cooling system specific activity.

II. ACCEPTANCE CRITERIA

Requirements

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

1. 10 CFR 20.1101, 10 CFR 20.1201, 10 CFR 20.1202, and 10 CFR 20.1206, as they relate to limiting occupational radiation doses.

2. 10 CFR 20.1203 and 10 CFR 20.1204, as they relate to limiting average concentrations of airborne radioactive materials to protect individuals and control the intake (inhalation or absorption) of such materials.

3. 10 CFR 20.1207, as it relates to limiting exposure to minors to one-tenth of limits for adults.

4. 10 CFR 20.1301 and 40 CFR Part 190, as it relates to the determination of radiation levels and radioactive materials concentrations within the components of the plant that could affect direct radiation exposure to members of the public.

5. 10 CFR 20.1801, as it relates to securing licensed materials against unauthorized removal.

6. General Design Criterion (GDC) 61 found in Appendix A to 10 CFR Part 50, as it relates to systems that may contain radioactive materials.

7. 10 CFR 50.34(f)(2)(vii), GDC 19, as they relate to the acceptable radiation conditions in the plant under accident conditions, and the source term release assumptions used to estimate calculate those conditions[1].

8. 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 DC has been constructed and will be operated in conformity with the DC, the provisions of the Atomic Energy Act (AEA), and the U.S. Nuclear Regulatory Commission's (NRC's) regulations.

9. 10 CFR 52.80(a), which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the COL, the provisions of the AEA, and the NRC's regulations.

10. 10 CFR 50.49(e)(4) and GDC 4, which requires the determination of the radiation environment expected during normal operation and the most severe design bases accident, for electric equipment relied upon to remain functional during and following design-basis events (DBEs), including AOOs.

11. 10 CFR 20.1406, as it relates to the identification of systems containing radioactive material for which the applicant should describe how the design minimizes contamination of the facility and environment, minimizes the generation of waste, and facilitates decommissioning.

12. 10 CFR 52.47(a)(22), as it relates to ensuring that information necessary to demonstrate how operating experience insights have been incorporated into the plant design.

13. 10 CFR 52.47(a)(5), 10 CFR 52.79(a)(3) and 10 CFR 52.157(e), as they relate to identifying the kinds and quantities of radioactive materials expected to be produced in

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[1]For Part 50 applicants not listed in 10 CFR 50.34(f), the provisions of 50.34(f) will be made a requirement during the licensing process.

the operation and the means for controlling and limiting radioactive effluents and radiation exposures within the limits set forth in Part 20 of this chapter.

DSRS Acceptance Criteria

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

The following regulatory guides (RGs), standards, and NUREGs provide information, recommendations, and guidance and in general describe a basis acceptable to the staff for implementing the requirements of 10 CFR 50.34(b)(3), 52.47(a)(5), 10 CFR 52.79(a)(3) and 10 CFR 52.157(e), 10 CFR 20.1201, 10 CFR 20.1202, 10 CFR 20.1203, 10 CFR 20.1204, 10 CFR 20.1206, 10 CFR 20.1207, 10 CFR 20.1301, 40 CFR Part 190, 10 CFR 20.1801, and 10 CFR 50.49.

1. RG 1.183[2], as it relates to the assumptions used in evaluating the concentrations of radionuclides in containment and plant systems following a loss-of-coolant accident.

2. RG 1.7, as it relates to radionuclides in systems used for determining gaseous concentrations in containment following an accident.

3. RG 1.112, as it relates to complying with the Commission's regulations under 10 CFR 20.1301 concerning the calculation of realistic radiation levels and radioactive materials source terms for the evaluation of waste treatment systems.

4. NUREG-0737, Task Action Plan Item II.B.2, using NuScale-specific source term values, as it relates to the identification of specific post accident sources of radiation in the facility.

5. American National Standards Institute/American Nuclear Society (ANSI/ANS) Standard 18.1, as it relates to methods and data used to estimate typical long-term concentrations of principal radionuclides in fluid streams of light-water-cooled nuclear power plants. These methods and data must be modified by the applicant to reflect NuScale-specific design attributes, which were developed using relevant industry operating experience.

6. RG 1.89 and RG 1.183, as they relate to the determination of radiation dose to certain electrical equipment important to safety as described in 10 CFR 50.49.

7. RG 1.143, regarding design features provided to minimize ORE and classification of structures housing radioactive waste systems based on potential exposure to site personnel.

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[2]RG 1.183 is applicable to applicants or license holders issued after January 10, 1997.

8. RG 1.26, RG 1.29 and RG 1.117, as they relate to the radiological criteria for classification and protection of non-radioactive waste SSCs containing radioactive material.

Compliance with the following specific acceptance criteria is necessary to meet the relevant requirements of the regulations identified above.

Descriptions should be provided for all radiation sources that require (1) shielding (e.g., provided for ORE 20.1101(b), 20.1201, 20.1202 or public 20.1301(e), radiation exposure control), (2) special ventilation systems, (3) special storage locations and conditions, (4) traffic or access control, (5) special plans or procedures, or (6) monitoring equipment. The source descriptions should include all pertinent information required for (1) input to shielding codes used in the design process, (2) establishment of related facility design features, (3) development of plans and procedures, (4) assessment of radiation (public or occupational, as appropriate) exposure, and (5) determination of radiation dose to electrical equipment important to safety as described in 10 CFR 50.49. Unless described within other sections of the FSAR, source descriptions should include the methods, models and assumptions used as the bases for all values provided in FSAR Section 12.2. A listing of isotope, quantity, form, and use of all required radiation sources containing byproduct, source, and special nuclear material exceeding 3.7E+9 Bq (100 millicuries) that may warrant shielding design consideration, should be provided.

For contained sources, the description should include plan scale drawings of each floor of the plant that show all sources identified so that they can easily be related to tables containing the pertinent and necessary quantitative source parameters. Their position should be located accurately, indicating the approximate size and shape. The information about neutron and gamma source terms from the core, reactor pressure vessel, irradiated components inside the reactor vessel, irradiated components inside containment, irradiated portions of the containment vessel, irradiated components and materials present outside of the containment vessel, and adjacent operating reactors, should be described in sufficient detail to allow determination of the radiation fields that could occur in areas that may require occupancy , may contain certain electrical equipment important to safety as described in 10 CFR 50.49 or could be a source of radioactive material available to systems outside of the containment vessel. Relevant experience from operating reactors may be used to the extent practicable. Airborne sources that are created by leakage, opening formerly closed containers, storage of leaking fuel elements, direct neutron activation of shielding materials and other plant materials, and other mechanisms should be identified by location and magnitude so that they can be used for designing appropriate ventilation systems and in specifying appropriate monitoring systems. Airborne radioactivity concentrations in frequently occupied areas should be a small fraction of the concentrations related to 10 CFR 20.1203, 10 CFR 20.1204, and Appendix B to 10 CFR Part 20. The assumptions made in arriving at quantitative values for these various sources should be specified, either in this section or by reference to FSAR Chapter 11, and other relevant interfacing sections of the FSAR.

Shielding and ventilation design fission product source terms will be acceptable if developed using these bases:

1. 0.25-percent fuel cladding defects for pressurized-water reactors (PWRs) or the reactor coolant system isotopic concentrations, including fission products and significant corrosion and activation products, equivalent to operation for a full fuel cycle at the technical specification limits for halogens (I-131 dose equivalent) and noble

gases (Xe-133 dose equivalent).

2. Post-accident shielding (for vital area access, including work in the area) source terms from RG 1.183, modified for NuScale-specific design attributes, which were developed using relevant industry operating experience.

3. Direct exposure to members of the public due to radiation from neutron activation products is less than the limits of 20.1301(e).

Coolant and corrosion activation products source terms should be based on applicable reactor operating experience. The buildup of activated corrosion products in various components and systems should be addressed. Any allowances made in design source terms for the buildup of activated corrosion products should be explained. Neutron and prompt gamma source terms should be based on reactor core physics calculations and applicable reactor operating experience.

Source terms for neutron activation of materials located outside of the containment vessel, or neutron activation of materials located inside of the reactor vessel that are rapidly transported outside of the containment vessel, should be based on reactor core physics calculations and the applicable reactor operating parameters.

The source term used for determining shielding and ventilation design of PWR components provided for purification of secondary coolant, should consider isotopic concentrations associated with operation at the Technical Specifications allowed limits for primary-to-secondary leakage and/or the secondary coolant specific activity concentrations, in addition to any direct neutron activation products.

The tables of source parameters, which can be placed in FSAR Chapter 12 or referenced to FSAR Chapter 11, will be acceptable if the accompanying text either in this section or other referenced sections makes it clear how the values are used in a shield design calculation, equipment qualification or in a ventilation system design during normal operation, AOOs as defined in 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," and DBEs as defined in Section (b)(ii) of 10 CFR 50.49, "Environmental qualification of electric equipment important to safety for nuclear power plants." The quantities of radioactive material occurring in systems connected to the reactor coolant system, and those materials resulting from primary to secondary leakage, will be acceptable if the specific values given in the tables are based on the methods presented in ANSI/ANS Standard 18.1 and RG 1.112, modified for NuScale-specific design attributes, which were developed using relevant industry operating experience, for coolant and corrosion activation products source terms. The types and quantities of radioactive material resulting from neutron activation of materials located outside of the containment vessel, or neutron activation of materials located inside of the reactor vessel that are rapidly transported outside of the containment vessel, should be tabularized in either FSAR Subsection 11.1 or in FSAR Subsection 12.2. For nuclear power plants designed for the recycling of tritiated water, tritium concentrations in contained sources and airborne concentrations in the regions specified in Item I.2 above should be based on a primary coolant concentration of 1.3E+5 Bq/gm (3.5 μCi/gm)), or an alternate value for which the methods, models, and assumptions have been provided in the application.

<u>Technical Rationale</u>

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. Compliance with the referenced sections of 10 CFR Part 20 requires that the licensee control both occupational dose and dose to individual members of the public from radioactivity that may be received from both internal and external sources. The licensee must also maintain the security of licensed radioactive materials that are stored in controlled or unrestricted areas.

2. Compliance with the referenced sections of 10 CFR Part 50 requires that the licensee control the radiation exposure to required plant equipment, and to personnel required to enter important areas, following DBEs.

3. Compliance with 10 CFR 52.47(a)(5), 10 CFR 52.79(a)(3) and 10 CFR 52.157(e) ensures that the kinds and quantities of radioactive materials expected to be produced in the operation are described so that the means for controlling and limiting radioactive effluents and radiation exposures within the limits set forth in Part 20 of this chapter can be identified.

Collectively, meeting these acceptance criteria ensures that all of the sources of radiation exposure to workers and members of the public resulting from the licensed activities (normal operations and AOOs) and to workers under accident conditions are identified, characterized, and considered in the design and operation of the facility, consistent with the relevant requirements of 10 CFR Part 20 and 10 CFR Part 50.

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.

The reviewer will consider whether source strengths, concentrations of airborne radioactivity, and quantitative source descriptions are consistent with the assumptions made and the methods used by the applicant. The reviewer should consider whether the bases for RCS source terms are consistent with relevant industry experience about erosion/corrosion rates, fuel integrity and the primary water chemistry control program (i.e., Electric Power Research Institute (EPRI) "Pressurized Water Reactor Primary Water Chemistry Guidelines"), including the High Duty Core Index. EPRI developed the "Advanced Light Water Reactor (ALWR) Utility Requirements Document Volume III, Passive Plants" (URD) based on proven technology of 40 years of commercial United States and international light-water reactor (LWR) experience. NUREG-1242, "NRC Review of Electric Power Research Institute's Advanced Light Water Reactor Utility Requirements Document," Volume 3, Part 1 and Volume 3, Part 2, documented the NRC staff's safety evaluation of the URD. The URD reviewed by the staff in 1992 referenced a number of industry documents, such as NP-6737, "Cobalt Reduction Guidelines," that provided contemporary operating experience regarding design practices beneficial to reducing ORE, by controlling potential sources of radiation. While the state of technology has advanced since the issuance of the initial URD, the reports referenced within the URD, revised versions of those reports and new reports (e.g., "Pressurized Water Reactor Primary Water Zinc Application Guidelines,") related to reducing the amount of radioactive material contained in

plant systems, are sources of information that describe the current state of technology that may be used to evaluate design specifications provided to ensure ORE is ALARA, consistent with the guidance in RG 8.8 and the requirements of 10 CFR 20.1003 and 1101(b), and 10 CFR 52.47(a)(22) to ensure that operating insights have been incorporated into the plant design. Consistent with the requirements of 10 CFR 52.47(a)(5), 10 CFR 52.79(a)(3), 52.137(a)(5) and 10 CFR 52.157(e), the staff should review how the applicant identifies the kinds and quantities of radioactive materials expected to be produced as a result of operating the facility, including fuel, irradiated reactor components, irradiation of other components within reactor vessels, irradiation of the containment vessels, and irradiation of the fluid surrounding the containment vessels. The staff should review how the applicant characterizes operating reactor/containment assemblies as sources of radiation to adjacent structures and personnel. The reviewer will examine locations of the contained sources relative to shield walls, occupied areas, traffic pathways, inservice inspection points, sampling stations, controls, and other parameters for special situations requiring additional action to ensure that ORE will be ALARA and that the bases for the radiation dose to equipment specified by 10 CFR 50.49 are described. The reviewer will consider sources of radioactive material that may be used to determine the qualification or required protective features, consistent with the radiological criteria described in RG 1.26, RG 1.29 and RG 1.117. Based on the review, the staff may request additional information or ask the applicant to reevaluate the analysis and modify those areas that do not meet the acceptance criteria given in Subsection II of this DSRS section.

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. 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 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 or other NRC approvals (e.g., manufacturing license, site suitability report or topical report).

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

IV. EVALUATION FINDINGS

The reviewer verifies that the applicant has provided sufficient information and that the staff's technical review and analysis support conclusions of the following type to be included in the staff's SER. The reviewer also states the bases for those conclusions.

1. The staff's review should verify that the FSAR and its amendments contain sufficient information, in accordance with the provisions of Section C.I.12.2 of RG 1.206 for DCs and COLs and 10 CFR 52.47 for DCs or 10 CFR 52.79 for COLs to arrive at conclusions of the following type, which are to be included in the staff's SER. The report will include

a summary of the applicant's submittal, the staff's basis for review and acceptance criteria, and the findings of the review. The following is a brief representation of the evaluation findings:

A. The staff concludes that the information provided by the applicant with respect to radiation sources is acceptable and meets the requirements of 10 CFR Part 20 and GDC 61 in Appendix A to 10 CFR Part 50. This conclusion is based on the following rationale:

B. The applicant has described a facility that can meet the requirements of 10 CFR 20.1101(b), 10 CFR 20.1201, 10 CFR 20.1202, 10 CFR 20.1203, 10 CFR 20.1204, 10 CFR 20.1206, 10 CFR 20.1207, 10 CFR 20.1301, 40 CFR Part 190, 10 CFR 20.1801, 10 CFR 20.1301(e), and 10 CFR 50.49, as they relate to the evaluation of source terms and the related provisions of GDCs 4, 19, and 61 in Appendix A to 10 CFR Part 50 and supplemented by the guidance of RG 1.112, RG 1.183, NUREG-0737 using the NuScale-specific source term, and ANSI/ANS Standard 18.1, modified for NuScale-specific design attributes, which were developed using relevant industry operating experience.

C. The applicant has provided a description of contained and airborne radioactivity sources used as inputs for the dose assessment and for shielding and ventilation designs. The applicant also included its assumptions in arriving at quantitative values for these contained and airborne source terms, based on ANSI/ANS-18.1, RG 1.112, GDC 61, 10 CFR 20.1201, 10 CFR 20.1202, 10 CFR 20.1203, 10 CFR 20.1204, 10 CFR 20.1206, and 10 CFR 20.1207 and Technical Specifications, modified for NuScale-specific design attributes, which were developed using relevant industry operating experience. For post-accident shielding for vital area access, the applicant used the source term in NUREG-0737 and RG 1.183, modified for NuScale-specific design attributes, which were developed using relevant industry operating experience.

During power operation, the greatest potential for personnel dose inside the reactor building from operating reactors are from nitrogen-16, noble gases, fission gammas and fission neutrons. After shutdown inside the reactor building, the primary sources of personnel exposure are fission products from fuel clad defects, activated components , activation products, including activated corrosion products and direct radiation from adjacent operating reactors. Outside of the reactor building (e.g., radioactive waste handling facilities) the primary sources of personnel exposure are fission products from fuel clad defects and activation products, including activated corrosion products. The coolant and corrosion activation product source terms are based on NuScale-specific design attributes, which were developed using relevant industry operating experience; allowances are included for the buildup of activated corrosion products. Neutron and prompt gamma source terms are based on reactor core physics calculations and NuScale-specific design attributes which were developed using relevant industry operating experience. Chapter 11 of the FSAR contains other parameters used, as well as a complete description of the routine operation source term development. The accident source terms are based on the guidance in RG 1.183, modified for NuScale-specific design attributes, which were developed using relevant industry operating experience. The bases for source terms presented for the NuScale-specific design are to the extent practicable, consistent with typical PWR design assumptions (i.e., construction materials, erosion rates, tramp contamination of fuel, etc.).

The applicant has provided a tabulation of the maximum expected routine radioactive airborne concentrations in equipment cubicles, corridors, and operating areas because of equipment leakage, or direct neutron activation of materials. The bases for the radioactivity from leakage calculations are in accordance with RG 1.112, modified for NuScale-specific design attributes, which were developed using relevant industry operating experience.

The bases for the source terms used to develop the airborne activity concentration values, due to corrosion and fission products, presented for the NuScale-specific design are to the extent practicable, consistent with typical PWR design assumptions (i.e., construction materials, erosion rates, pressure boundary leakage, etc.), and NuScale-specific design attributes. Due to NuScale-specific neutron shielding provisions, the staff should review the source terms and assumptions associated with the neutron activation of the reactor shield pool, and the air above operating reactors.

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.”