U.S. NUCLEAR REGULATORY COMMISSION

DESIGN-SPECIFIC REVIEW STANDARD for NuScale SMR DESIGN

12.3–12.4 RADIATION PROTECTION DESIGN FEATURES

REVIEW RESPONSIBILITIES

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

Secondary - None

I. AREAS OF REVIEW

The U.S. Nuclear Regulatory Commission (NRC) staff will review the applicant's final safety analysis report (FSAR) for a design certification (DC) or combined license (COL). The staff reviews radiation protection design features, taking into account design dose rates, anticipated operational occurrences (AOOs), and accident conditions. The staff will review radiation protection design features to ensure the applicant's design reflects occupational radiation exposure (ORE) from direct and airborne radioactive material in the facility as low as is reasonably achievable (ALARA), controls the exposure of members of the public to direct radiation from sources located at the facility, minimizes contamination of the facility and the environment, minimizes the generation of waste, and protects equipment important to safety.

The specific areas of review are as follows:

1. Facility Design Features

   A. In the DC FSAR or the COL FSAR, the description of equipment and facility design features used for assuring that ORE will be ALARA consistent with Title 10 of the *Code of Federal Regulations* (10 CFR), Section 20.1101(b), and 10 CFR 20, Subpart H, "Respiratory Protection and Controls to Restrict Internal Exposure in Restricted Areas."

   B. The radiation zone designations, including zone boundaries for normal operational (including abnormal operational occurrences), refueling, and accident conditions (based on Regulatory Guides (RGs) 1.7, "Control of Combustible Gas Concentrations in Containment," and 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,") (review based on descriptions provided in the DC FSAR or the COL FSAR to the extent that the information is not addressed in a referenced certified design).

   C. The illustrative examples of facility design features of the equipment, components, and systems, including clearly readable scaled layout and arrangement drawings of the facility showing all source locations and the other design details, as requested in Section C.I.12.3.1 of RG 1.206, "Combined License Applications for Nuclear Power Plants," (review based on descriptions provided in the DC FSAR or the COL FSAR to the extent that the information is not addressed in a referenced certified design);

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specification of shield wall thicknesses for all shielded spaces on drawings or in separate tables.

D. Information describing the implementation of the guidelines in RG 8.8, “Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable,” contained within regulatory position C.2 “Facility and Equipment Design Features,” on facility and equipment design and layout, as well as information describing any proposed alternatives provided to maintain ORE ALARA (review based on descriptions provided in the DC FSAR or the COL FSAR to the extent that the information is not addressed in a referenced certified design).

E. Information describing design features that will facilitate eventual decommissioning and minimize, to the extent practicable, contamination of the facility and environment and the generation of radioactive waste in accordance with 10 CFR 20.1406, “Minimization of Contamination,” (DC FSAR or the COL FSAR to the extent that they are not addressed in a referenced certified design). The guidance contained in RG 4.21, “Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning,” and Appendix 12.3-12.4-A, “Evaluation and Acceptance Criteria for 10 CFR 20.1406 to Support Design Certification and Combined License Applications,” enclosed herein, is intended to be used by reviewers of this design-specific review standard (DSRS) section, and other Standard Review Plan (SRP) or DSRS section reviewers, as applicable.

F. Information describing the implementation of RG 8.8 guidelines to reduce the production, distribution, and retention of activation products through the specification for materials and features of components that will be in direct contact with primary coolant, or provide information describing any proposed alternatives consistent with the requirements of 10 CFR 20.1001(b) and 10 CFR 52.47(a)(22) (review based on descriptions provided in the DC FSAR or the COL FSAR to the extent that the information is not addressed in a referenced certified design).

G. Information describing the implementation of RG 8.8 guidelines on the specifications for equipment and components provided to improve reliability, reduce leakage from systems containing radioactive material, facilitate maintenance and reduce required inspections, as well as information describing any proposed alternatives (review based on descriptions provided in the DC FSAR or the COL FSAR to the extent that the information is not addressed in a referenced certified design), consistent with the requirements of 10 CFR 20.1001(b) and 10 CFR 52.47(a)(22).

H. Information describing the implementation of RG 8.8 guidelines on the specifications for station lighting features to provide a favorable working environment, promote work efficiency, and facilitate egress from high radiation areas if the station lighting system fails, through the use adequate lighting, the use of extended service lamps, design features that permit the servicing of the lamps from lower radiation areas and provisions for emergency lighting, as well as information describing any proposed alternatives consistent with the requirements of 10 CFR 20.1001(b) and 10 CFR 52.47(a)(22) (review based on descriptions provided in the DC FSAR or the COL FSAR to the extent that the information is not addressed in a referenced certified design).

I. Information describing the implementation of RG 8.8 guidelines on the description and location of each very high radiation area on plant layout diagrams and the design features provided to control access to radiologically restricted areas (including potentially very high radiation areas), such as the reactor cavity and the fuel transfer tube during refueling operations, as well as information describing any proposed alternatives consistent with the requirements of 10 CFR 20.1001(b) and 10 CFR 52.47(a)(22) (review based on descriptions provided in the DC FSAR or the COL FSAR to the extent that the information is not addressed in a referenced certified design).

2. <u>Shielding</u>

A. The shielding to be provided for each of the radiation sources identified in FSAR Chapter 11 and Section 12.2, and other applicable sections, including the design criteria and the shield material to be used for penetrations, and to preclude radiation (including neutron) streaming into containment or other areas that may be occupied (review based on descriptions provided in the DC FSAR or the COL FSAR to the extent that they are not addressed in a referenced certified design); specification of shield wall thicknesses for all shielded spaces on the plant layout drawings or in separate tables (as noted in Item I.1.C above).

B. The description of the methods by which the shield parameters were determined, including pertinent codes, assumptions, and techniques used or to be used in the calculations (review based on descriptions provided in the DC FSAR or the COL FSAR to the extent that the information is not addressed in a referenced certified design).

C. The description of any special protective features that use shielding, geometric arrangement, or remote handling that will ensure that ORE will be ALARA (review based on descriptions provided in the DC FSAR or the COL FSAR to the extent that they are not addressed in a referenced certified design).

D. Information describing implementation of RGs 1.69, "Concrete Radiation Shields and Generic Shield Testing for Nuclear Power Plants," and 8.8 (regarding special protective features), and information describing any proposed alternatives (review based on descriptions provided in the DC FSAR or the COL FSAR to the extent that they are not addressed in a referenced certified design).

E. Descriptions and location of areas (including the access to and egress from) that personnel may need to access following an accident (as required by 10 CFR 50.34(f)(2)(vii) and the guidance contained in NUREG-0737, Item II.B.2), using NuScale-specific source term calculations (review based on descriptions provided in the DC FSAR or the COL FSAR to the extent that the information is not addressed in a referenced certified design).

F. Physical layout and composition of plant structures and walls that provide shielding for, and barriers to, high and very high radiation areas such that personnel access to and work within these areas can be controlled in accordance with 10 CFR Part 20 Subpart F, “Surveys and Monitoring,” consistent with the guidance contained within RG 8.38, “Control of Access to High and Very High Radiation Areas in Nuclear Power Plants” (review based on descriptions provided in the DC FSAR or the COL FSAR to the extent that the information is not addressed in a referenced certified design).

3. Ventilation

A. The description of the personnel protection features incorporated in the ventilation system designs provided in Section C.I.12.3.3 of RG 1.206 (review based on descriptions provided in the DC FSAR or the COL FSAR to the extent that the information is not addressed in a referenced certified design).

B. Illustrative examples of personnel radiation protection features of the air cleaning system design (review based on descriptions provided in the DC FSAR or the COL FSAR to the extent that the information is not addressed in a referenced certified design).

C. Information describing the application of the guidance contained within RG 1.52, “Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants”; RG 1.140, “Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants”; RG 8.15, “Acceptable Programs for Respiratory Protection”; and RG 8.8, regarding design features provided to minimize ORE from exposure from airborne radioactive material, and to minimize ORE because of operation and maintenance of the equipment (review based on descriptions provided in the DC FSAR or the COL FSAR to the extent that the information is not addressed in a referenced certified design).

4. Area Radiation and Airborne Radioactivity Monitoring Instrumentation

A. The description of the fixed area radiation and continuous airborne radioactivity monitoring instrumentation for normal operation, AOOs, and accident conditions, including the criteria for placement, called for in Section C.I.12.3.4 of RG 1.206 (review based on descriptions provided in the DC FSAR or the COL FSAR to the extent that the information is not addressed in a referenced certified design).

B. The criteria and method for obtaining representative in-plant airborne radioactivity concentrations in work areas (review based on descriptions provided in the DC FSAR or the COL FSAR to the extent that the information is not addressed in a referenced certified design).

C. Description of procedures for locating suspected high-activity radioactivity areas.

D. Information describing the implementation of radiation monitoring equipment criteria listed in RGs 8.2, “Administrative Practices in Radiation Surveys and Monitoring”; 8.8; 8.25, “Air Sampling in the Workplace”; and 1.97, “Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants”; and American National Standards Institute/Health Physics Society (ANSI/HPS)

Standard N13.1-2011, and information describing any proposed alternatives (review based on descriptions provided in the DC FSAR or the COL FSAR to the extent that they are not addressed in a referenced certified design).

E. Utilizing the NuScale-specific source term, the description of safety-related radiation monitoring capability after an accident, such as: the in-containment high-range monitors required by 10 CFR 50.34(f)(2)(xvii), as discussed in the guidance of Item II.F.1.3 of NUREG 0737, the guidance of RG 1.97, and DSRS Chapter 7, and the requirements of 10 CFR Part 50 Appendix E VI.2(a)(i) for radiation monitors that are part of the emergency response data system (ERDS).

F. Utilizing the NuScale-specific source term, the description of in-plant radiation airborne radioactivity monitoring system in accordance with 10 CFR 50.34(f)(2)(xxvii), as discussed in the guidance of Item II.F.1.3 of NUREG-0737, the guidance of RG 1.97, and DSRS Chapter 7.

G. Description of locations for fixed radiation monitors in accordance with ANSI/American Nuclear Society (ANS)/Health Physics Society Standards Committee (HPSSC) Standard ANSI/ANS/HPSSC-6.8.1.

H. Description of radiation monitors in areas where special nuclear material is handled or stored in accordance with 10 CFR 50.68, or 10 CFR 70.24, both titled “Criticality Accident Requirements,”.

I. RG 1.97 and DSRS Section 11.6, “Guidance on Instrumentation and Control Design Features for Process and Effluent Radiological Monitoring, and Area Radiation and Airborne Radioactivity Monitoring,” as they relate to compliance with NRC regulations that provide instrumentation to monitor important-to-safety plant variables and systems during and following an accident.

5. <u>Dose Assessment</u>

A. The description of the basis for the dose assessment process, providing detailed information as to expected occupancy of plant radiation areas for each radiation zone, and the estimated annual person-sievert (person-rem) doses associated with major functions, such as operation, Radwaste handling, normal maintenance, special maintenance (e.g., steam generator tube plugging), refueling, and inservice inspection, in accordance with the provisions of RG 8.19, “Occupational Radiation Dose Assessment in Light-Water Reactor Power Plants Design Stage Man-Rem Estimates” (DC FSAR or COL FSAR).

B. The description of any additional dose-reducing measures taken because of the dose assessment process for specific functions or activities (review based on descriptions provided in the DC FSAR or the COL FSAR to the extent that the information is not addressed in a referenced certified design).

C. For facilities being constructed adjacent to an existing operating nuclear unit(s), in accordance with the guidance contained in RG 1.206, Subsection C.I.12.3.5 and the guidance contained in NUREG-1555, provision of a description of the basis for the dose assessment process for plant construction workers, providing detailed information as to the estimated number of construction workers and estimated annual doses (from direct, gaseous, and liquid sources) to these workers, in accordance with the provisions of NUREG-1555 (review based on descriptions provided in the DC FSAR or the COL FSAR to the extent that they are not described in a referenced certified design).

6. 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 DSRS section in accordance with SRP Section 14.3, "Inspections, Tests, Analyses, and Acceptance Criteria." The staff recognizes that the review of ITAAC cannot be completed until after the rest of this portion of the application has been reviewed against acceptance criteria contained in this DSRS section. Furthermore, the staff reviews the ITAAC to ensure that all SSCs in this area of review are identified and addressed as appropriate in accordance with SRP Section 14.3.

7. 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 Technical Submittal 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 radiation protection design feature information provided in the Technical Submittal for the specific plant design:

1. Section 3.7.4, "Seismic Instrumentation," as it relates to the station layout and design features provided to reduce ORE dose to personnel who operate, service, or inspect seismic instrumentation, and specifications for seismic instruments that include consideration for long service life and low frequency of maintenance for the expected radiation conditions.

2. Section 3.8.2, "Steel Containment," as it relates to the material specifications and radiation protection design features provided: to reduce ORE; to control direct dose to members of the public; to support inservice inspections of structures; and to minimize contamination, to the extent practicable, of the facility or environment.

3. Section 3.8.3, "Concrete and Steel Internal Structures of Steel or Concrete Containments," as it relates to the radiation protection design features provided: to reduce ORE; to protect plant equipment and to control direct dose to members of the public; to protect areas that may contain irradiated fuel or irradiated components; to protect access points to very high radiation areas; for describing structural shielding materials, including dimensions and specifications for materials used for shielding; to support inservice inspections of structures; and to minimize contamination, to the extent practicable, of the facility or environment.

4. Section 3.8.4, “Other Seismic Category I Structures,” as it relates to the radiation protection design features: provided to reduce ORE and protect plant equipment; in areas that may contain irradiated fuel or irradiated components; for access points to very high radiation areas; of structural shielding materials, including dimensions and specifications for materials used for shielding; needed to support inservice inspections of structures; used to minimize contamination, to the extent practicable, of the facility or environment.

5. Section 3.11, “Environmental Qualification of Mechanical and Electrical Equipment,” as it relates to design features provided to control radiation exposure in order to maintain the qualification of mechanical, electrical, and electronic equipment.

6. Section 4.2, “Fuel System Design,” as it relates to design features provided to minimize ORE from the fuel, such as cladding material and grid straps.

7. Section 4.3 “Nuclear Design,” as it relates to the specification of materials to reduce ORE, such as specifications for low cobalt content, the use of ORE reducing technologies provided for testing and maintenance, the design features provided to minimize contamination from testing and maintenance activities, and the methods to reduce the quantity and volume of radioactive waste.

8. Section 4.5.1, “Control Rod Drive Structural Materials,” as it relates to the neutron absorber materials and fabrication design criteria, provided to reduce ORE.

9. Section 4.5.2, “Reactor Internal and Core Support Structure Materials,” as it relates to the specification of materials to reduce ORE, such as specifications for low cobalt content; the use of ORE reducing technologies, such as zinc injection; the types and methods of construction (e.g., the use of double wall pins versus single walled pins to improve integrity) of startup neutron sources; the types of neutron detection equipment (e.g., in core neutron detectors); and features provided to minimize neutron irradiation of plant structures and components.

10. Section 5.2.3, “Reactor Coolant Pressure Boundary Materials,” as it relates to the material specifications provided to reduce ORE (e.g., low cobalt content); chemistry controls to minimize corrosion and reduce ORE (e.g., Electric Power Research Institute (EPRI) primary water chemistry guidelines and zinc injection); thermal hydraulic design features provided to limit erosion (e.g., limiting flow rates, or the use of baffles); and fabrication techniques, such as processes to ensure smooth surfaces resistant to erosion or the deposition of material.

11. Section 5.2.4, “Reactor Coolant Pressure Boundary Inservice Inspection and Testing,” as it relates to the methods and features provided to reduce ORE because of inspections such as reducing inspection frequencies, improving access to SSCs, and using improved inspection techniques.

12. Section 5.2.5, “Reactor Coolant Pressure Boundary Leakage Detection,” as it relates to the radiation monitoring systems, types of detectors and specified sensitivity provided for leakage detection to ensure the integrity of the reactor coolant system (RCS), and to minimize contamination of the facility and reduce ORE through early detection of leaks.

13. Section 5.3.1, “Reactor Vessel Materials,” as it relates to the material specifications provided to reduce ORE (e.g., cobalt content).

14. Section 5.4, “Reactor Coolant System Component and Subsystem Design,” as it relates to the features provided to limit or reduce the buildup of radioactivity in tanks, heat exchangers, and related components connected to the RCS, and features provided to limit ORE minimize contamination of the facility and reduce waste generation from these potential radiation sources.

15. Section 5.4.2.1, “Steam Generator Materials,” as it relates to the materials (e.g., cobalt content) and design features (e.g., access for tube testing) provided to minimize ORE minimize contamination of the facility and facilitate decommissioning.

16. Section 5.4.7, “Residual Heat Removal (RHR) System,” as it relates to the design features provided to minimize introduction of corrosion products into the RCS, to minimize ORE from activity contents of the system, and to minimize contamination of the facility and minimize leakage of radioactive fluids.

17. Section 5.4.2.2, “Steam Generator Program,” (or the equivalent NuScale system) as it relates to the sensitivity and detector types specified for radiation monitors provided for compliance with Nuclear Energy Institute (NEI) 97-06 and the underlying EPRI Guidelines, and to minimize contamination of the facility and reduce ORE through early detection of leaks.

18. Section 6.1.1, “Engineered Safety Features Materials,” as it relates to the design features provided to minimize introduction of corrosion products into the RCS, to minimize ORE from activity contents of the system, and to minimize contamination of the facility and minimize leakage of radioactive fluids.

19. Section 6.1.2, “Protective Coating Systems (Paints)—Organic Materials,” as it relates to the qualification of and application of coatings provided for minimizing contamination of the facility and environment, and to the design features provided to minimize the ORE associated with the inspection and maintenance of those coatings.

20. Section 6.2.1, “Containment System Functional Design,” as it relates to the design features provided to minimize ORE because of operation, inspection, and maintenance of containment SSCs.

21. Section 6.2.2, “Containment Heat Removal Systems,” as it relates to the design features provided to minimize ORE because of operation, inspection, and maintenance of containment heat removal SSCs; to maintain ORE to personnel accessing containment ALARA; and to control radiation exposure in order to maintain the qualification of mechanical, electrical, and electronic equipment.

22. 6.2.8 Containment Evacuation System (or the equivalent NuScale system,) as it relates to design features provided to minimize ORE from the system during operation and maintenance; to minimize contamination of the facility and minimize leakage of radioactive fluids; and to the extent that the instrumentation credited for RCS leakage detection is described in FSAR Chapter 12, the types of detectors and the bases for the specified sensitivity.

23. Section 6.3, “Emergency Core Cooling System,” as it relates to design features provided for leakage detection and prevention; to minimize ORE from the system during operation and maintenance and to minimize contamination of the facility and leakage of radioactive fluids.

24. Section 6.4, “Control Room Habitability System,” as it relates to the design features provided to minimize ORE because of operation, testing, and maintenance of control room ventilation system SSCs; to shield operators from radiation exposure; and to protect control room operators from airborne radioactive materials.

25. Section 6.5.3, “Fission Product Control Systems and Structures,” as it relates to limiting or reducing radioactive fission, activation and corrosion product sources within the SSCs; features provided to reduce ORE during operation, testing, and maintenance of Engineered Safety Features (ESF) SSCs; and to provisions to minimize contamination of the facility and minimize leakage of radioactive fluids.

26. Section 6.6, “Inservice Inspection of Class 2 and 3 Components,” as it relates to the methods and features provided to reduce ORE because of inspections such as reducing inspection frequencies, improving access to SSCs, and using improved inspection techniques.

27. Chapter 7, Appendix B, “Instrumentation and Controls System Architecture,” as it relates to the design features of safety-related equipment that provide for radiological protection of plant workers, reducing ORE associated with servicing and maintaining of plant instrumentation, and to minimize contamination of the facility and the environment.

28. Section 7.1, “Fundamental Design Principles,” as it relates to the design features of safety-related equipment that provide for radiological protection of plant workers, reducing ORE associated with servicing and maintaining of equipment.

29. Section 7.2, “System Characteristics,” as it relates to the safety-related instrumentation provided for monitoring radiological conditions during an accident.

30. Section 11.6, “Guidance on Instrumentation and Control Design Features for Process and Effluent Radiological Monitoring, and Area Radiation and Airborne Radioactivity Monitoring,” as it relates to compliance with NRC regulations to provide instrumentation to monitor important-to-safety plant variables and systems during and following an accident. This includes the types, ranges, and qualification of radiation monitoring equipment required for accident monitoring, and design features provided for radiological protection of plant workers, reducing ORE associated with servicing and maintaining of plant instrumentation and minimizing contamination of the facility and the environment.

31. Section 9.1.1, “Criticality Safety of Fresh and Spent Fuel Storage and Handling,” as it relates to the features provided to ensure adequate shielding and cooling of irradiated fuel and irradiated core components, within the refueling area, in transit and in spent fuel pool storage areas in order to maintain ORE ALARA.

32. Section 9.1.2 “New and Spent Fuel Storage,” as it relates to the features provided to ensure adequate shielding and radiation monitoring of new fuel, irradiated fuel, and irradiated components, within the refueling area, in transit and in storage or maintenance areas in order to maintain ORE ALARA.

33. Section 9.1.3, “Spent Fuel Pool Cooling and Cleanup System Handling,” as it relates to features provided to ensure adequate shielding and cooling of irradiated fuel and components within the refueling area, in transit and in spent fuel pool storage areas; features provided to reduce ORE from the radioactive material resulting from direct neutron activation of the contents of the reactor module pool fluid; shielding and design features of filtration and purification media of refueling and fuel storage pools provided to maintain ORE ALARA; features provided to minimize contamination of the facility and the environment; and features provided to detect and minimize leakage of radioactive fluids from system components.

34. Section 9.1.4, “Light Load Handling System (Related to Refueling),” as it relates to the design features provided to ensure adequate shielding during storage, movement, and handling of spent fuel and irradiated components in order to maintain ORE ALARA.

35. Section 9.1.5, “Overhead Heavy Load Handling Systems,” as it relates to the design features provided to ensure adequate shielding during reactor operation and storage, movement, and handling of spent fuel and irradiated components, in order to maintain ORE ALARA.

36. Section 9.2.2, “Reactor Auxiliary Cooling Water System,” as it relates to the design features provided to minimize and reduce the amount of radioactive fission, activation and corrosion products contained within the system; reduce ORE during operation testing and maintenance; and minimize contamination of the facility and minimize leakage of radioactive fluids.

37. Section 9.2.4, “Potable and Sanitary Water Systems,” as it relates to the design features provided to minimize and reduce the amount of radioactive fission, activation and corrosion products contained within the system in order to minimize contamination of the facility and minimize leakage of radioactive fluids.

38. Section 9.2.6, “Condensate Storage Facilities,” as it relates to the design features provided to minimize and reduce the amount of radioactive fission, activation, and corrosion products contained within the system, including those resulting from direct neutron activation of the contents of the secondary fluid systems contained within the reactor vessel; to reduce ORE during operation testing and maintenance; and to minimize contamination of the facility and minimize leakage of radioactive fluids and control exposure of members of the public to direct sources of radiation.

39. Section 9.3.1, “Compressed Air System,” as it relates to design features provided to prevent radiological contamination of the system. Also, if the system is used as a source of breathing air for radiological protection, the design features provided to ensure protection of personnel, including protection from contaminants, air quality monitoring, and provisions for ensuring adequate air supply to respiratory protection equipment users.

40. Section 9.3.2, “Process and Post-Accident Sampling Systems,” as it relates to design features provided to minimize ORE during operation and AOOs and design-basis events (DBEs); minimize waste generation; obtain representative samples; minimize contamination of the facility; and minimize leakage of highly radioactive fluids.

41. Section 9.3.3, “Equipment and Floor Drainage System,” as it relates to design features provided to minimize and remove sources of radiation (e.g., crud traps), and to minimize contamination of the facility and the environment.

42. Section 9.3.6, “Auxiliary and Radwaste Area Ventilation System,” as it relates to limiting or reducing radioactive fission, activation, and corrosion product sources within the ventilation filtration media, tanks and structures; providing features to reduce ORE during operation testing and maintenance of ventilation systems components; and providing features to minimize contamination of the facility.

43. Section 9.3.4 Chemical and Volume Control System (PWR) (including Boron Recovery System), as it relates to the minimization, reduction, and shielding of radioactive fission, activation, and corrosion products within the system’s piping tanks and vessels, including the associated filtration and purification media and features provided to minimize the introduction of material into the RCS, such as the specification of low cobalt content; equipment design features that limit erosion (e.g., smooth surfaces or design flow rates and baffles to help reduce erosion); features provided to minimize system leakage; and features provided to minimize required maintenance.

44. Section 9.4.1, “Control Room Area Ventilation System,” as it relates to limiting or reducing radioactive fission, activation, and corrosion product sources within the ventilation filtration media, tanks and structures; providing features to reduce ORE during operation testing and maintenance of ventilation systems components; and providing features to minimize contamination of the facility.

45. Section 9.4.2, “Spent Fuel Pool Area Ventilation System,” as it relates to limiting or reducing radioactive fission, activation and corrosion product sources within the ventilation filtration media, tanks and structures; providing features to reduce ORE during operation testing and maintenance of ventilation systems components; and providing features to minimize contamination of the facility.

46. Section 9.4.3, “Auxiliary and Radwaste Area Ventilation System,” as it relates to limiting or reducing radioactive fission, activation and corrosion product sources within the ventilation filtration media, tanks and structures; providing features to reduce ORE during operation because of evaporation from pools, venting of containment vessels, venting of tanks or process components, or the testing and maintenance of ventilation systems components; and providing features to minimize contamination of the facility.

47. Section 9.5.2, “Communications Systems,” as it relates to ensuring adequate communications are provided for the purpose of minimizing ORE within the radiologically controlled area during operation, AOOs, and DBEs.

48. Section 9.5.3, “Lighting Systems,” as it relates to design features for providing adequate normal and emergency lighting in radiologically controlled areas during operation, AOOs, and DBEs; and providing features to minimize ORE associated with maintenance and servicing of normal and emergency lighting.

49. Section 10.2, “Turbine Generator,” as it relates to limiting or reducing the content of activation and corrosion products resulting from direct neutron activation of the contents of the secondary fluid systems contained within the reactor vessel; and to providing features to reduce ORE during operation, testing, and maintenance of ventilation

systems components, providing features to minimize contamination of the facility, and limiting direct radiation exposure to members of the public.

50. Section 10.3, “Main Steam Supply System,” as it relates to limiting or reducing the content of activation and corrosion products resulting from direct neutron activation of the contents of the secondary fluid systems contained within the reactor vessel; providing features to reduce ORE during operation, testing, and maintenance of ventilation systems components; providing features to minimize contamination of the facility; and limiting direct radiation exposure to members of the public.

51. Section 10.4.1, “Main Condensers,” as it relates to limiting or reducing the content of activation and corrosion products resulting from direct neutron activation of the contents of the secondary fluid systems contained within the reactor vessel; providing features to reduce ORE during operation, testing and maintenance of ventilation systems components; providing features to minimize contamination of the facility; and limiting direct radiation exposure to members of the public.

52. Section 10.4.6, “Condensate Cleanup System,” as it relates to limiting and reducing radioactive fission, activation, and corrosion products within the system piping tanks and vessels, including the associated filtration and purification media; providing design features to limit ORE during operation, testing, and maintenance; and providing design features to minimize contamination of the facility.

53. Section 10.4.8, “Steam Generator Blowdown System” (or the equivalent NuScale system), as it relates to limiting and reducing radioactive fission, activation, and corrosion products within the system piping tanks and vessels, including the associated filtration and purification media; providing design features to limit ORE during operation, testing, and maintenance; and providing design features to minimize contamination of the facility.

54. Section 10.4.11, “Auxiliary Boiler System” (or the equivalent NuScale system), as it relates to the design features provided to minimize contamination of the facility and environment, facilitate decommissioning, and minimize the generation of radioactive waste.

55. Section 10.4.12, “Decay Heat Removal System” (or the equivalent NuScale system), as it relates to providing design features to limit ORE during inspection, testing, and maintenance, and the design features provided to limit the potential for airborne radioactive material during inspection, testing, and maintenance.

56. Chapter 11, “Radioactive Waste Management,” as it relates to the description of the design features provided for containment, shielding, and handling of material contained within the radioactive waste management system; provisions for maintaining ORE ALARA during routine operation, AOOs and DBEs; and providing design features to minimize contamination of the facility.

57. Chapter 14, “Radiation Protection—Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC),” as it relates to ITAAC for radiation protection equipment and equipment provided to reduce ORE.

58. Chapter 16, “Technical Specifications,” as it relates to identification of requirements for High Radiation Area and very high radiation area access controls, and any requirements for radiation monitors described in Chapter 12.

II. ACCEPTANCE CRITERIA

Requirements

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

1. 10 CFR 20.1101(b) and the definition of ALARA in 10 CFR 20.1003, “Definitions,” as they relate to persons involved in licensed activities making every reasonable effort, and engineering controls to maintain radiation exposures ALARA.

2. 10 CFR 20.1201, “Occupational Dose Limits for Adults,” as it relates to occupational dose limits for adults.

3. 10 CFR 20.1201; 10 CFR 20.1202, “Compliance with Requirements for Summation of External and Internal Doses”; 10 CFR 20.1203, “Determination of External Dose from Airborne Radioactive Material”; 10 CFR 20.1204, “Determination of Internal Exposure”; 10 CFR 20.1701, “Use of Process or Other Engineering Controls”; and 10 CFR 20.1702, “Use of Other Controls,” as they relate to design features, ventilation, monitoring, and dose assessment for controlling the intake of radioactive materials.

4. 10 CFR 20.1301, “Dose Limits for Individual Members of the Public,” and 10 CFR 20.1302, “Compliance with Dose Limits for Individual Members of the Public,” as they relate to the facility design features that affect the radiation exposure to a member of the public from non-effluent sources associated with normal operations and AOOs.

5. 10 CFR 20.1406, as it relates to the design features that will facilitate eventual decommissioning and minimize, to the extent practicable, the contamination of the facility and the generation of radioactive waste.

6. 10 CFR 20.1601, “Control of Access to High Radiation Areas”; 10 CFR 20.1602, “Control of Access to Very High Radiation Areas”; 10 CFR 20.1901, “Caution Signs”; 10 CFR 20.1902, “Posting Requirements”; 10 CFR 20.1903, “Exceptions to Posting Requirements”; and 10 CFR 20.1904, “Labeling Containers”; as they relate to the identification of potential sources of radiation exposure and the controls of access to and work within areas of the facility with a high potential for radiation exposure.

7. 10 CFR 20.1801, “Storage and Control of Licensed Material,” as it relates to securing licensed materials against unauthorized removal from the place of storage.

8. General Design Criterion (GDC) 19, “Control Room,” found in Appendix A to 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities,” as it relates to the provision of adequate radiation protection to permit access to areas necessary for occupancy after an accident, without personnel receiving radiation exposures in excess of the 50 millisievert (mSv) (5 rem) total effective dose equivalent (TEDE), as defined in 10 CFR 50.2, “Definitions,” to the whole body or the equivalent to any part of the whole body for the duration of the accident in accordance with

10 CFR 50.34(f)(2)(vii)[1] and NUREG-0737, Item II.B.2, using the NuScale-specific source term.

9. GDC 61, "Fuel Storage and Handling and Radioactivity Control," as it relates to occupational radiation protection aspects of fuel storage, handling, radioactive waste, and other systems that may contain radioactivity, designed to ensure adequate safety during normal and postulated accident conditions, with suitable shielding and appropriate containment and filtering systems.

10. GDC 63, "Monitoring Fuel and Waste Storage," as it relates to detecting excessive radiation levels in the facility.

11. 10 CFR 50.68 or 10 CFR 70.24, as it relates to procedures and criteria for radiation monitoring in areas where special nuclear material is stored and handled.

12. 10 CFR 52.47(b)(1), which requires that a DC FSAR contain the 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 NRC regulations.

13. 10 CFR 52.80(a), which requires that a COL FSAR 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 NRC regulations.

14. 10 CFR 50.34(f)(2)(xvii) and NUREG-0737, Item II.F.1, using NuScale-specific source term which requires the applicant to provide instrumentation to monitor containment radiation intensity (high level).

15. 10 CFR 50.49(e)(4) and GDC 4, which require the determination of the radiation environment expected during normal operation and the most severe design basis accident, for electric equipment relied upon to remain functional during and following DBEs, including AOOs.

16. 10 CFR 50.34(f)(2)(vii), Section II.B.2 of NUREG-0737, using the NuScale-specific source term, which requires the performance of radiation shielding design reviews to ensure the design permits adequate access to important areas and provides for protection of safety equipment from radiation, following an accident.

17. GDC 14, "Reactor Coolant Pressure Boundary," and GDC 30, "Quality of Reactor Coolant Pressure Boundary," as they relate to the ability to detect RCS pressure boundary leakage with radiation detectors.

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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.

18. 10 CFR Part 50, Appendix E, “Emergency Planning and Preparedness for Production and Utilization Facilities,” Section VI.2(a)(i), which requires radiation monitoring systems for reactor coolant radioactivity, containment radiation level, condenser air removal radiation level, and process radiation monitor levels.

19. 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.

20. 10 CFR 50.34(b)(3), 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 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.

<u>DSRS Acceptance Criteria</u>

Specific DSRS acceptance criteria acceptable to meet the relevant requirements of NRC regulations identified above are set forth below. The DSRS is not a substitute for NRC 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 NRC regulations that underlie the DSRS acceptance criteria.

The following RGs, NUREGs, and industry standards provide information, recommendations, and guidance and, in general, describe a basis acceptable to the staff for implementing the requirements of the regulations identified above.

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

2. RG 1.52, as it relates to radiation protection considerations for ESF atmosphere cleanup systems operable under postulated design-basis accident (DBA) conditions, to be designated as “primary systems.”

3. RG 1.69, as it relates to the requirements and recommended practices acceptable for construction of facilities that apply to occupational radiation protection shielding structures for nuclear power plants.

4. RG 1.97, DSRS Chapter 7, and the Memorandum from D.G. Eisenhut, NRR, to Regional Administrators dated August 16, 1982, as they relate to methods acceptable to the staff for complying with NRC regulations to provide and calibrate, or verify calibration of, safety-related instrumentation for radiation monitoring following an accident in a nuclear power plant.

5. RG 1.183[2], as it relates to the assumptions and methods for evaluating doses to individuals accessing the facility during and following an accident in accordance with NUREG-0737, Item II.B.2.

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

6. RG 8.2, as it relates to general information on radiation monitoring programs for administrative personnel.

7. RG 8.8, as it relates to actions taken during facility design, engineering, construction, operation, and decommissioning to maintain ORE ALARA in accordance with 10 CFR 20.1101(b) and the definition of ALARA in 10 CFR 20.1003, concerning the radiation protection information to be supplied in FSAR Section 12.

8. RG 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable," as it relates to the commitment by management and vigilance by the radiation protection manager and staff to maintain ORE ALARA in accordance with 10 CFR 20.1101(b) and the definition of ALARA in 10 CFR 20.1003.

9. RG 8.15, as it relates to methods acceptable to the staff for ensuring the safety of personnel utilizing an installed breathing air system, provided for radiological respiratory protection.

10. RG 8.19, as it relates to a method acceptable to the staff for performing an assessment of collective occupational radiation dose as part of the ongoing design review process so that such exposures will be ALARA.

11. RG 8.25, as it relates to a method acceptable to the staff for continuous monitoring for airborne radioactive materials in plant spaces.

12. RG 8.38, as it relates to the physical controls for personnel access to high and very high radiation areas.

13. ANSI/ANS/HPSSC-6.8.1-1981, "Location and Design Criteria for Area Radiation Monitoring Systems for Light Water Nuclear Reactors," as it relates to criteria for the establishment of locations for fixed continuous area gamma radiation monitors and for design features and ranges of measurement.

14. ANSI/HPS N13.1-2011, "Sampling and Monitoring Releases of Airborne Radioactive Substances from the Stacks and Ducts of Nuclear Facilities," as it relates to the principles that apply in obtaining valid samples of airborne radioactive materials, and acceptable methods and materials for gas and particle sampling.

15. ANSI/ANS-6.4-2006, "Nuclear Analysis and Design of Concrete Radiation Shielding for Nuclear Power Plants," as it relates to requirements and recommended practices for the construction of concrete radiation shielding structures.

16. Memorandum from Larry W. Camper to David B. Matthews and Elmo E. Collins, dated October 10, 2006, and NUREG/CR-3587, "Identification and Evaluation of Facilitation Techniques for Decommissioning Light Water Power Reactors" as they relate to the design issues that need to be addressed to meet the requirements of 10 CFR 20.1406.

17. RG 1.140, as it relates to actions taken to address the guidance contained in RG 8.8, Position C.2(d), during facility design, engineering, construction, and decommissioning to maintain ORE ALARA in accordance with 10 CFR 20.1101(b) and the definition of ALARA in 10 CFR 20.1003, concerning the radiation protection information to be supplied in FSAR Section 12.

18. RG 1.89, “Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants,” as it relates to the determination of radiation dose to certain electrical equipment important to safety as described in 10 CFR 50.49, “Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants.”

19. RG 4.21, as it relates to the design features provided to minimize contamination of the facility and environment, facilitate decommissioning, and minimize the generation of radioactive waste.

20. RG 1.45, “Guidance on Monitoring and Responding to Reactor Coolant System Leakage,” as it relates to the detection capabilities of radiation monitors described in Chapter 12 that are provided for RCS pressure boundary leakage detection, to the extent that they are not addressed in other sections of the DSRS.

21. NEI 97-06, “Steam Generator Program Guidelines,” as it relates to the leakage detection capabilities of the radiation monitoring equipment described in Chapter 12 of the FSAR, that are provided to detect steam generator tube leakage, in accordance with the criteria specified in the EPRI bases documents to the extent that they are not addressed in other sections of the DSRS.

22. RG 1.143, “Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants,” regarding design features provided to minimize ORE and classification of structures housing radioactive waste systems based on potential exposure to site personnel.

23. Branch Technical Position (BTP) 11-3, “Design Guidance for Solid Radioactive Waste Management Systems Installed in Light-Water-Cooled Nuclear Power Reactor Plants,” and SECY-94-198, “Review of Existing Guidance Concerning the Extended Storage of Low-Level Radioactive Waste,” as they relate to design features provided to minimize ORE for radioactive waste storage facilities described in the application.

24. RG 1.97, “Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants,” NUREG-0800, DSRS Section 11.6, “Guidance on Instrumentation and Control Design Features for Process and Effluent Radiological Monitoring, and Area Radiation and Airborne Radioactivity Monitoring,” and the Memorandum from D.G. Eisenhut, Nuclear Reactor Regulation, to Regional Administrators dated August 16, 1982, as it relates to a method acceptable to the staff for complying with the NRC’s regulations that require the licensee to provide and calibrate radiation monitoring instrumentation, as they relate to monitoring important to safety plant variables and systems during and following an accident.

25. RG 1.12, “Nuclear Power Plant Instrumentation for Earthquakes,” as it relates to minimizing ORE through the selection of locations for installing seismic monitoring equipment, and the selection of equipment design specifications that reduce the frequency or duration of testing, inspection, or maintenance of seismic monitoring equipment.

The specific acceptance criteria are:

i. Facility Design Features

The acceptability of the facility design features will be based on evidence that the applicant has fulfilled the dose-limiting requirements of 10 CFR 50.34(b)(3); 10 CFR 52.47(a)(5); 10 CFR 52.79(a)(3); 10 CFR 52.157(e); 10 CFR 20.1201; 10 CFR 20.1202; 10 CFR 20.1203; 10 CFR 20.1204; 10 CFR 20.1207, “Occupational Dose Limits for Minors”; 10 CFR Part 20, Subpart H, “Respiratory Protection and Controls to Restrict Internal Exposure in Restricted Areas”; 10 CFR 20.1301; and 10 CFR 20.1302; as well as the radiation protection aspects of GDCs 19 and 61, and 10 CFR 50.34, “Contents of Applications; Technical Information,” and 10 CFR 50.49 for controlling radiation dose to electrical equipment important to safety and 10 CFR 52.47(a)(22). This includes evidence that major exposure accumulating functions (maintenance, refueling, radioactive material handling and processing, inservice inspection, calibration, decommissioning, and recovery from accidents) have been considered in plant design. The evidence should also include radiation protection features incorporated into the design, taking into account the state of technology, to keep potential radiation exposure from these activities ALARA in accordance with 10 CFR 20.1101(b), the definition of ALARA in 10 CFR 20.1003, and RGs 8.8 and 8.10.

Such features may include (1) the ease of accessibility to work, inspection, and sampling areas, (2) the ability to reduce source intensity, (3) design measures to reduce the production, distribution, and retention of activated corrosion products (e.g., material selection, water chemistry and decontamination connections), including those resulting from direct neutron activation, (4) the ability to reduce time required in radiation fields, (5) a provision for portable shielding and remote handling tools, (6) protection from neutron and gamma radiation from operating reactors, and (7) control of and ORE protection from direct neutron activation products (solid, liquid and gaseous) produced outside of the containment vessel. Evidence of methods to control personnel exposure from high dose rate components such as temporary storage areas for irradiated fuel and irradiated core components (e.g., storage and handling of fixed in core detectors during outage), irradiated containment vessels, steam generator tubes and contaminated heat exchangers, are considered during plant design. Access control will be judged for acceptability in accordance with the requirements of 10 CFR 20.1601, 10 CFR 20.1602, 10 CFR 20.1901, 10 CFR 20.1902, and 10 CFR 20.1903, or access control alternatives in the NuScale Technical Specifications. Evidence of methods to control radiation exposure to members of the public from all sources of direct radiation (i.e., non-effluent), including direct neutron activation products, has been considered during plant design, and limits dose from all sources of direct radiation in accordance with the requirements of 10 CFR 20.1301(e). Where the design includes provisions for the supply of breathing air for radiological protection of personnel, evidence of methods to ensure breathing air quality must be consistent with the requirements of 10 CFR 20.1703(g).

Facility design, to the extent practicable, should minimize the potential for creating a very high radiation area during normal operations, including AOOs (such as dropping a fuel bundle during fuel handling operations). High and very high radiation areas should be remote from normally occupied rooms and corridors such that personnel access to these areas can be controlled in accordance with 10 CFR 20.1601, 10 CFR 20.1602 and the guidance in RG 8.38. All accessible portions of the facility that are capable of having radiation levels greater than 1 gray (Gy) per hour (100 rads per hour) should be shielded. This shielding should be such that the resultant contact radiation levels are no greater than 1 Gy per hour (100 rads per hour). All accessible portions of the facility capable of having radiation levels greater than 1 Gy per hour (100 rads per hour) are clearly marked with a sign stating that potentially lethal radiation fields are possible. If removable shielding is used to reduce dose rates to less than 1 Gy per hour, it must also be explicitly marked as above. If other than permanent shielding is used, local audible and visible alarming radiation monitors must be installed to alert personnel if temporary shielding is removed during operations that may result in dose rates greater than 1 Gy per hour. Consistent with 10 CFR 20.1602, facility design should ensure that an individual is not able to gain unauthorized or inadvertent access to areas in which radiation levels could be encountered at 500 rads (5 grays) or more in 1 hour at 1 meter from a radiation source or any surface through which the radiation penetrates (e.g., those adjacent to operating reactors, or irradiated portions or reactor or containment vessels, of shut down reactors).

The areas inside the plant structures, as well as in the general plant yard, should be subdivided into radiation zones, with maximum design dose rate zones and the criteria used in selecting maximum dose rates identified. Maximum zone dose rates should be defined for each zone, depending on anticipated occupancy and access control. The areas that must be occupied on a predictable basis (based on the number of people and stay or transit times) during normal operations (including refueling; purging; fuel handling and storage; radioactive material handling; processing, use, storage, and disposal; normal maintenance; routine operational surveillance; inservice inspection; and calibration) and AOOs should be zoned such that this occupancy results in an annual dose to each of the involved individuals that is as far below the limits of 10 CFR Part 20 as is reasonably achievable, and a total person-sievert (person-rem) dose that is ALARA. Based on current operating experience and on predictions being made for new plant designs, it is expected that the plant shielding can be designed, the plant can be zoned, and sufficient radiation protection design features can be incorporated, such that individuals in shielded areas would receive a small fraction of the 10 CFR Part 20 limits.

All vital areas, in which radiation may unduly limit personnel occupancy during operations following an accident resulting in a degraded core, should be identified. Personnel access to these areas under accident conditions should be demonstrated in accordance with 10 CFR 50.34(f)(2)(vii), using the methods listed in Section II.B.2 of NUREG-0737, using the NuScale-specific source term. The analysis should consider access to, stay time in, and egress from these vital areas. All vital areas, in which radiation may unduly limit personnel occupancy during operations following a loss of consumable or variable shielding (e.g., water) around operating reactors or shutdown reactors containing irradiated fuel, should be identified. Personnel access to these areas under accident conditions should be demonstrated in accordance with 10 CFR 20, Subpart C, “Occupational Dose Limits.” The analysis should consider access to, stay time in, and egress from these vital areas.

Consistent with the guidance contained in RGs 8.8 and 1.143, BTP 11-3 and SECY-94-198, “Review of Existing Guidance Concerning the Extended Storage of Low-Level Radioactive Waste,” SSCs that are described in the application, should be designed to control leakage and facilitate access, operation, inspection, testing, and maintenance in order to maintain radiation exposures to operating and maintenance personnel ALARA. Structures housing radioactive waste processing systems or components should be classified using the guidance for potential exposure to site personnel found in RG 1.143.

ii. Shielding

The staff will evaluate the shielding design in terms of the assumptions used to calculate shield thickness, the calculational methods used, and the parameters chosen. A number of acceptable shielding calculational codes are available that are effective for determining the necessary shield thickness for gamma ray and combination neutron-gamma sources. The code description file of the Radiation Safety Information Computational Center (formerly the Radiation Shielding Information Center) at Oak Ridge National Laboratory includes most of the codes used by shield designers, which means that the codes have been tested and authenticated for operation but not for reliability and accuracy. Radiation shielding codes vary in complexity and accuracy from the relatively simple point-kernel methods, to the more complex discrete ordinates methods, to the still more rigorous Monte Carlo methods. The staff may use these codes, as necessary, to calculate dose rates for given shield designs and source strengths as a confirmation of the applicant’s method.

The applicant’s methods for performing shielding design calculations are acceptable if the methods are comparable to commonly accepted shielding calculations; if assumptions regarding source terms, cross sections, shield and source geometries, and transport methods are realistic; and if specified radiation zones are consistent with the assumed source term and shielding specified in the design. Labyrinth shielded access ways and penetrations should be used to minimize radiation streaming and scatter around shields. Composition of the shielding material should be selected to minimize, to the extent practicable, the potential for the shield itself to become a radiation source (either from activation of the shield material or production of secondary radiation resulting from interactions with the primary radiation).

Where the applicant’s shielding design incorporates consumable or variable material, to provide shielding for areas in which radiation levels could be encountered at 500 rads (5 grays) or more in 1 hour at 1 meter from a radiation source or any surface through which the radiation penetrates (e.g., those adjacent to operating reactors, or irradiated portions or reactor or containment vessels of shut down reactors), the applicant should demonstrate how the requirements of 10 CFR 20.1601 and 10 CFR 20.1602 will be met for all conditions of normal operations, AOOs, and DBEs. Effective shield design is essential to meeting the criteria that ORE will be ALARA, the requirements of

10 CFR 20, Subpart C, for plant workers will be met, the requirements of 10 CFR 20.1602 for plant workers will be met, and the requirements of 10 CFR 20.1301(e) will be met for members of the public. Accordingly, the applicant should describe how the facility design will provide reasonable assurance of adequate protection of the public health and safety, and that the plant can be operated in conformity with NRC rules and regulations for those conditions where consumable or variable shielding material is absent.

Where the applicant's shielding design incorporates material subject to degradation, such as through the effects of radiation (e.g., depletion of boron neutron absorbers), temperature extremes (e.g., degradation of polymer based materials because of high temperature), density changes (e.g., sagging or settling of shielding material with age), the reviewer should ensure that methods are in place to ensure that ORE remains ALARA, and the equipment exposures are maintained in accordance with the provisions of 10 CFR 50.49. The staff should review how the application identifies the allowable constraints (e.g., minimum cooling air flow, maximum shielding material temperature, and maximum allowable neutron flux), and how those parameters are measured and assessed over the design life of the facility. Note that in some cases shielding integrity may be required through decommissioning.

In addition, RG 1.69 and ANSI/ANS-6.4-2006 provide guidance on the fabrication and installation of concrete shields for occupational radiation protection at nuclear power plants. Acceptability of the shield construction will be based on an indication that the guidance of these documents have been implemented in facility construction, or that acceptable alternatives have been proposed. RG 8.8 provides additional acceptance criteria regarding shielding and isolation in radiation protection design.

iii. <u>Ventilation</u>

The ventilation system will be acceptable for radiation protection purposes if the criteria and bases for ventilation rates within the plant will ensure that air will flow from areas of low potential airborne radioactivity to areas of higher airborne radioactivity and then to filters or vents; that the concentrations of radioactive material in areas normally occupied can be maintained in accordance with the requirements 10 CFR 20.1701; and that the dose limits of 10 CFR 20.1201 are met consistent with the requirements of 10 CFR 20.1202, 10 CFR 20.1203, and 10 CFR 20.1204. Where physical barriers are not present between areas of potentially higher and lower concentrations of airborne radioactivity, the application describes the methods, models, and assumptions used to derive the assumed air flow patterns utilized by the applicant to assess the adequacy of the ventilation flow directions, determination of supply and return register locations, and fixed continuous airborne radioactivity sample points. The system has adequate capability to reduce concentrations of airborne radioactivity to 1.0 derived air concentration (DAC), as specified in Appendix B to 10 CFR Part 20, in areas not normally occupied where maintenance or inservice inspection must be performed. The system is designed so that filters containing radioactivity can be easily maintained and will not create an additional radiation hazard to personnel maintaining them, or those in adjacent occupied areas, consistent with the guidance contained in RG 8.8 and the requirements of 10 CFR 20.1101(b) and

10 CFR 52.47(a)(22). Acceptability of the ventilation system, relative to radioactive gases and particulates, will also be based on evidence that the applicant has applied the guidance of RG 8.8 and RG 1.140 or proposed acceptable alternatives.

For components, such as heat exchangers that are normally submerged in contaminated water during operation, the reviewer should determine what design features will be provided to minimize and mitigate airborne contamination from these components when they are exposed to air during refueling.

RG 1.52, particularly Sections C.3.10 and 4.10, provides guidance that can be used in this review, although the guide relates to mitigating accidents involving airborne radioactivity. Good practices in this regard apply to normal operation as well, since the release of radioactivity in normal operational occurrences is usually different only in quantity from some of the accident cases.

iv. <u>Area Radiation and Airborne Radioactivity Monitoring Systems</u>

A. The area radiation monitoring systems will be acceptable if they meet the provisions of 10 CFR 20.1501, "General," 10 CFR 50.34(f)(2)(xvii), the guidance in NUREG-0737 using the NuScale-specific source term, RG 8.25, RG 1.97, DSRS Chapter 7 for safety-related systems, or DSRS Section 11.5, DSRS Section 11.6, "Guidance on Instrumentation and Control Design Features for Process and Effluent Radiological Monitoring, and Area Radiation and Airborne Radioactivity Monitoring," and the following criteria:

i. The detectors are located in areas that normally may be occupied without restricted access and that may have a potential for radiation fields in excess of the radiation zone designations discussed in the third paragraph under Item 1, above, in accordance with ANSI/ANS-HPSSC-6.8.1.

ii. The detectors provide on-scale readings of dose rate that include the design maximum dose rate of the radiation zone in which they are located, as well as the maximum dose rate for AOOs and accidents.

iii. The detectors are calibrated during fuel outages and after the performance of any maintenance work on the detector.

iv. Each monitor has a local audible alarm and variable alarm set points. Monitors located in high noise areas should also have visual alarms.

v. Readout and annunciation are provided in the control room.

vi. The in-containment high-range radiation monitors meet the criteria of 10 CFR 50.34(f)(2)(xvii).

vii. Emergency power is initiated after a loss of offsite power.

B. The airborne radioactivity monitoring system will be acceptable if it is consistent with the guidance on continuous air sampling in RG 8.25 and meets the following criteria:

i. Engineering controls provide the principal protection against the intake of radioactive materials.

ii. Air should be sampled at normally occupied locations where airborne radioactivity may exist, such as solid waste handling areas, spent fuel pools, and reactor operating floors. The monitoring system should be capable of detecting 10 DAC-hours of particulate and iodine radioactivity from any compartment that has a possibility of containing airborne radioactivity and that normally may be occupied by personnel, taking into account dilution in the ventilation system. Continuous monitoring of air being exhausted from locations within the facility during normal operation is an acceptable method. Noble gas monitors should be calibrated such that, when monitoring for ^{133}Xe, the instrument response will determine concentrations accurately.

iii. Representative air concentrations are measured at the detectors, which are located as close as possible to the sampler intakes.

iv. Ventilation monitors are upstream of high-efficiency particulate air filters.

v. The detectors are calibrated routinely and after any maintenance work is performed on the detector.

vi. Each location has a local audible alarm and variable alarm set points. Monitors located in high noise areas should also have visual alarms.

vii. Readout and annunciation are provided in the control room.

viii. Emergency power is initiated after a loss of offsite power.

C. The in-plant accident radiation monitoring systems will be acceptable if they meet the following criteria:

i. Personnel have the capability to assess the radiation hazard in areas that may be accessed during the course of an accident, in accordance with the criteria of 10 CFR 50.34(f)(2)(xvii); NUREG-0737, Item II.F.1; RG 1.97, using the NuScale-specific source term, with appropriate margins; DSRS Section 11.6, “Guidance on Instrumentation and Control Design Features for Process and Effluent Radiological Monitoring, and Area Radiation and Airborne Radioactivity Monitoring”; and the Memorandum from D.G. Eisenhut, NRR, to Regional Administrators dated August 16, 1982, about calibration of radiation monitoring equipment.

ii. Portable instruments to be used in the event of an accident should be placed to be readily available to personnel responding to an emergency.

iii. Emergency power should be provided for installed accident monitoring systems.

iv. The accident monitoring systems should have usable ranges that include the maximum calculated accident levels and should be designed to operate properly in the environment caused by the accident.

v. Two high-range radiation monitors are provided to measure dose rates inside containment in accordance with the requirements of 10 CFR 50.34(f)(2)(xvii) and Item II.F.1 of NUREG-0737, using the NuScale-specific source term, with appropriate margins.

vi. For normal operations, AOOs and DBEs that may result in decreased quantities of consumable or variable shielding around operating, or shutdown reactors containing irradiated fuel, the installed radiation monitoring instruments provided should have ranges that include the maximum calculated dose rates postulated for a loss of shielding, consistent with the requirement of GDC 63 to detect excessive radiation levels.

vii. The application should describe the methods available for plant operators to assess the radiological conditions around shutdown reactors containing irradiated fuel, and adjacent to permanent shielding (e.g., the outside of the pool concrete shielding) following a total loss of consumable or variable shielding.

D. Appendix A to RG 1.21 provides useful guidance about effluent monitoring that applies to the acceptability of in-plant airborne radioactivity monitoring. RG 8.2 includes guidance on surveys to evaluate radiation hazards. The detailed guidance in ANSI N13.1-2011 covers the sampling of airborne radioactive materials in ventilation ducts and stacks of nuclear facilities and may be used for acceptance criteria on the actual sampling process and certain techniques involved. RG 8.8 provides further guidance on monitoring systems.

E. Instrumentation for monitoring areas where reactor fuel is stored or handled will be acceptable if it meets the criteria of 10 CFR 50.68 or 10 CFR 70.24.

F. To the extent that it is not covered in Section 5.2.5 or Section 11, the specified sensitivity of radiation monitoring equipment provided for reactor coolant pressure boundary leakage detection (i.e., RG 1.45) is acceptable if it is capable of meeting the required fluid leakage detection criteria.

G. To the extent that it is not covered in BTP 5-1 or Section 11.5, the specified sensitivity of radiation monitoring equipment provided for primary to secondary Leakage detection (i.e., requirements specified in the EPRI guidance forming the basis for NEI 97-06) is acceptable if it is capable of meeting the required fluid leakage detection criteria.

H. To the extent that it is not covered in Section 11.5, a description of the Emergency Response Data System (ERDS) radiation monitoring system components required by 10 CFR Part 50, Appendix E, Section VI.2(a)(i).

NUREG-1394 provides the minimum standards and acceptable methods that may be used to implement and comply with the ERDS requirements.

1. Dose Assessment

   The dose assessment will be acceptable if it documents the assumptions made, calculations used, results for each radiation zone (including numbers and types of workers involved in each), expected and design dose rates, and projected annual person-Sievert (person-rem) doses, in accordance with RG 8.19.

   If applicable, the applicant's dose assessment of construction workers on a facility adjacent to an existing nuclear unit(s), will be acceptable if it documents the assumptions made, calculations used, results for the areas where construction workers will be located (including numbers of construction workers), expected dose contributions (from direct, gaseous, and liquid sources), and projected person-Sievert (person-rem) doses, consistent with RG 1.206 Subsection C.I.12.3.5.

2. Minimization of Contamination

   Compliance with 10 CFR 20.1406 requires the applicant to describe how facility design and procedures for operation will minimize, to the extent practicable, contamination of the facility and the environment, facilitate eventual decommissioning, and minimize, to the extent practicable, the generation of radioactive waste. The acceptability of these features will be based on the guidance contained in RG 4.21 and Appendix 12.3-12.4-A (DC FSAR or the COL FSAR to the extent that they are not addressed in a referenced certified design.)

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. The referenced sections of 10 CFR Part 20 specify that the licensee shall control the radiation sources and the radiation doses to workers and members of the public from exposures to these sources, during normal operations, AOOs, and decommissioning, so that they are within the regulatory dose limits and ALARA.

2. The referenced sections of 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," and the associated sections of RG 1.206, specify the scope of the material in an application and the associated technical review by the staff.

3. The references to the specific items in 10 CFR 50.34(f), their associated action item in NUREG-0737, using the NuScale-specific source term, RG 1.97 and DSRS Chapter 7 for safety-related systems, or DSRS Section 11.6, "Guidance on Instrumentation and Control Design Features for Process and Effluent Radiological Monitoring, and Area Radiation and Airborne Radioactivity Monitoring," specify that adequate in-plant radiation monitoring is provided for accidents and abnormal operational occurrences. Radiation protection design features are provided to allow personnel access to the plant under accident conditions sufficient to perform actions necessary to mitigate the consequences of the accident.

4. Compliance with GDC 61 requires that systems that may contain radioactivity be designed to ensure adequate safety under normal and postulated accident conditions. This criterion specifies that such facilities shall be designed with appropriate containment, confinement, and filtering systems.

   The requirements of this GDC apply to DSRS Section 12.3-12.4 because systems and components that contain radioactive material are a potential source of radiation exposure to individual workers in the event of leakage of the systems or components, during normal operation, AOOs, or in the event of an accident.

   Meeting the requirements of GDC 61 provides a level of assurance that releases of radioactive materials during normal operation and AOOs will not result in radiation doses that exceed the limits specified in 10 CFR Part 20. In addition, meeting the requirements will help ensure that systems continue to perform safety functions under postulated accident conditions.

5. Compliance with the requirements of 10 CFR 50.68 or 10 CFR 70.24 and GDC 63 ensures that appropriate radiation monitoring is provided in areas of the plant where special nuclear material is handled, used, or stored. In addition, GDC 63 provides for adequate monitoring spaces containing radioactive waste systems. Prompt detection of excessive radiation levels in these areas resulting from normal operations or abnormal operational occurrences is necessary to identify potentially hazardous conditions for the plant workers and possible releases of radioactivity.

6. Compliance with 10 CFR 50.49(e)(4) and GDC 4 ensures that the radiation environment expected during normal operation and the most severe design bases accident will not exceed the functional capabilities of electric equipment relied upon to remain functional during and following DBEs, including AOOs.

7. Compliance with the requirement of GDC 30 ensures that means are provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

8. Compliance with the requirements of 10 CFR Part 50, Appendix E, Section VI.2(a)(i), ensures the provision of accurate and timely data needed to determine core and coolant system conditions well enough to assess the extent or likelihood of core damage and to determine the conditions inside the containment vessel well enough to assess the likelihood and consequence of its failure.

9. Compliance with the requirements of 10 CFR 20.1406 in an early stage of planning ensures that the facility will be designed and operated, to the extent practicable, in a way that would minimize the contamination of the facility, contamination of the environment, and the generation of radioactive waste, and would facilitate decommissioning. 10 CFR 20.1406 applies to all DC and COL applications submitted after August 20, 1997.

10. Compliance with 10 CFR 50.34(b)(3), 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.

III. REVIEW PROCEDURES

Several aspects of the radiation protection design features of the NuScale SMR design differ from those traditionally found in PWRs. The reactor building contains up to 12 reactors that are each enclosed in a separate containment vessel, which are submerged in a common pool of water that is used as the ultimate heat sink (UHS). The UHS, instead of concrete, provides the primary shielding (i.e., the shielding immediately around the reactor vessel). During refueling, the containment vessel, including the contained reactor vessel and all fuel for that reactor, is moved as an integral unit to the attached refueling pool. In the refueling portion of the pool, the containment and reactor vessels are disassembled and fuel is removed and placed in the attached spent fuel pool. The three connected pools (the reactor cooling pool, the refueling pool, and the spent fuel pool) are interconnected and enclosed by a single reactor building. While one reactor is being refueled, up to 11 other reactors, located in the contiguous reactor cooling pool, may continue to operate. The steam generators utilize helical coil tubes, with the secondary coolant on the inside of the tubes, and reactor coolant on the outside of the tubes. Control rods, and the associated drive mechanisms, are all fully contained within the reactor vessel. The decay heat removal heat exchanger, which is mounted on the exterior of the containment vessel, is normally submerged in the joint reactor shielding/refueling pool, and is subject to drying during refueling. Spent resin and liquid waste storage tanks and related processing equipment are in a separate building located adjacent to the reactor building.

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

1. Selected Programs and Guidance—In accordance with the guidance in NUREG–0800, "Introduction – Part 2: Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: Light-Water Small Modular 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 small modular reactor (SMR) framework. Using the graded approach described in NUREG-0800, Intro Part 2, the NRC staff may determine that, for certain 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 completing the applicable ITAAC. The use of the selected programs to augment or replace traditional review procedures is shown in Figure 1 of NUREG-0800, Intro Part 2. Examples of such programs that may be relevant to the graded approach for these SSCs include:

   - 10 CFR Part 50, Appendix A, GDC, Overall Requirements, Criteria 1–5
   - 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 Nonsafety 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 DC or COL applications submitted under 10 CFR 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, “Resolution of Generic Safety Issues,” 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 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 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). Reference 10 CFR 52.47(a)(21), 10 CFR 52.47(a)(22), and 10 CFR 52.47(a)(8), respectively. 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.

4. The staff will review the information on radiation protection design features furnished in the FSAR, including referenced parts of Chapters 3, 4, 5, 6, 7, 9, 10, 11, 13.4, 14.3.8, and 16, for completeness in accordance with RG 1.206 for DC or COL applicants under 10 CFR Part 52. The reviewer will evaluate the FSAR text and the scaled layout drawings of the facility, concentrating on:

- the sources, shielding, and layouts for the reactor building, and auxiliary building including the radwaste systems,

- containment vessel transfer SSCs,

- fuel handling SSCs, including the spent fuel pool fuel transfer and related equipment and temporary fuel handling or storage locations,

- irradiated component handling and storage locations,

- other areas containing radioactive material or contributing to the radioactive content of plant systems, and

- decontamination facilities, office and access control areas, laundry, lockers and shower rooms (including the personnel decontamination area), and laboratory facilities.

When available, the staff should review how the selection of the location of and specifications of plant monitoring equipment (e.g., seismic detectors, and loose parts monitoring instruments) reduces ORE. The review will evaluate the radiation protection design features using the guidelines of RG 8.8 and system specific guidance, such as RG 1.12. EPRI developed the “Advanced Light Water Reactor Utility Requirements Document,” Volume 3, ALWR Passive Plants,” based on proven technology of 40 years of commercial U.S. and international LWR experience. NUREG-1242, “NRC Review of Electric Power Research Institute’s Advanced Light Water Reactor Utility Requirements Document,” Volume 3, Parts 1 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-6516, “Guide for the Application and Use of Valves in Power Plant Systems,” NP-5479, “Application Guidelines for Check Valves in Nuclear Power Plants,” NP-5697, “Valve Stem Packing Improvements,” NP-6737, “Cobalt Reduction Guidelines,” and NP-6316, “Guidelines for Threaded-Fastener Application in Nuclear Power Plants,” that provided contemporary operating experience regarding design practices beneficial to reducing ORE. 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 improving equipment reliability 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 through the use of reliable and low maintenance valves, pumps, and other components, 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. The reviewer will consider plant layout and intended access and egress traffic patterns both to determine conformance with 10 CFR 20.1601, 10 CFR 20.1602, 10 CFR 20.1901, 10 CFR 20.1902, 10 CFR 20.1903, 10 CFR 20.1904, and

10 CFR 20.1905, or the applicant's Technical Specifications, and to determine whether they will control access properly in limited and restricted access areas (high radiation and very high radiation areas). The staff will review FSAR Chapters 5, 9 and 11, as necessary to evaluate dose rates in and around the spent fuel pool areas, the location of airborne radioactivity monitoring instruments within ventilation systems, and radwaste systems as they relate to radiation protection design. The reviewer will evaluate the methods, models, and assumptions utilized by the applicant to derive the assumed air flow patterns used to assess the adequacy of the ventilation flow directions and determine supply and return register locations and fixed continuous airborne radioactivity sample points in large areas where physical barriers are not present between areas of potentially higher and lower concentrations of airborne radioactivity. The information evaluated by the staff should address potential radiological conditions expected to result from normal operations, refueling outages, and AOOs, and the performance of the ventilation system. The Information contained in NUREG-1400, "Air Sampling in the Workplace," may be used to inform the staff's review of these design features. The reviewer will evaluate all relevant aspects of the initial design plans, particularly to identify new arrangements, improved designs, unusual shield thicknesses, a new or modified shield thickness calculational procedure, unusual assumptions in the calculation, and placement of radiation monitors. The staff responsible for the review of DSRS Chapter 11 will evaluate the adequacy of the process and effluent radiation monitoring (e.g., sensitivity, range, system placement) design.

5. RG 1.97 and DSRS Chapter 7 for safety-related systems, or DSRS Section 11.5, DSRS Section 11.6, "Guidance on Instrumentation and Control Design Features for Process and Effluent Radiological Monitoring, and Area Radiation and Airborne Radioactivity Monitoring," provide detailed guidance and criteria for postaccident radiation monitoring instrumentation. The staff will coordinate the review of the radiation monitoring systems with the instrumentation and control and emergency preparedness review staff to ensure that adequate radiation detection instrumentation is provided for plant monitoring under accident conditions.

6. The health physics staff will evaluate the adequacy of the applicant's shielding design on the basis of acceptable radiation shielding practices and calculation methods. The primary shielding for fission neutrons and gammas from the operating reactors is provided by the reactor pool, which is common for containment vessel cooling, refueling and spent fuel pools. Based on its review of the plant layout drawings and radiation zoning, the health physics staff may verify, by independent calculations, the adequacy of the shielding design for selected areas of the plant. The review should emphasize areas in the plant that have a potential to become a significant high radiation area (greater than 1 Gy (100 rads per hour) or a very high radiation area during operations and AOOs. These areas include, but are not limited to, areas providing access to the spent fuel during containment transfer, areas adjacent to the irradiated portions of the reactor vessel, and the containment vessel. The staff will evaluate the design features provided to prevent personnel from gaining unauthorized or inadvertent access to significant high radiation areas through the movement of material, the removal of consumable or variable shielding, or the removal of shielding blocks provided to facilitate plant maintenance. Appendix B to RG 8.38 includes guidance on some of these areas. The staff will determine that the applicant has identified all of the vital areas in which radiation may unduly limit personnel occupancy during operations following a loss of consumable or variable shielding around operating reactors or shutdown reactors containing irradiated fuel. The staff will evaluate how personnel access to these areas under

accident conditions is assured. The analysis should consider access to, stay-time in, and egress from all vital areas. The analysis should consider the potential impact to other operating reactors and reactors being refueled or maintained.

7. The reviewer will determine whether the applicant has followed the guidance of the referenced RGs and industry standards, both by comparison of the applicant's methods with the information in the guides and by the applicant's reference to any such guides or to proposed alternatives. The reviewer will evaluate whether the alternatives are equivalent to, or improvements on, the methods cited in the referenced RGs. Otherwise, alternatives are likely to be disapproved.

8. Based on the review, the health physics staff may request additional information or request the applicant to reevaluate the radiation protection design features to meet the acceptance criteria of Subsection II of this DSRS section.

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

10. 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.

11. The staff will review the information on design features furnished in the FSAR provided to minimize contamination of the facility and environment and minimize the generation of radioactive waste and facilitate decommissioning, including applicable parts of SRP and DSRS Chapters 3, 4, 5, 6, 7, 9, 10, 11, 16 SRP 13.4, 14.3.8, for completeness in accordance with the guidance contained in RG 4.21 and Appendix 12.3-12.4-A (or RG 1.206 for DC or COL applicants under 10 CFR Part 52). The reviewer will evaluate the FSAR text and the scaled layout drawings of the facility for descriptions of the design features provided to minimize contamination and facilitate decommissioning.

12. Using the guidance contained in Appendix 12.3-12.4-A and RG 4.21, the staff will review information in the application provided to describe how facility design and procedures for operation will minimize, to the extent practicable, contamination of the facility and the environment, facilitate eventual decommissioning, and minimize, to the extent practicable, the generation of radioactive waste. The reviewer will use the information provided in Chapter 12 of the FSAR, supplemented and complemented as necessary by information contained in those sections of the FSAR that describe systems containing radioactive material during normal operations, AOOs, and accident conditions.

13. For any portion of the post-accident subsystems that support safety-related functions, as identified by the applicant, the review of these design features is performed under DSRS

Chapter 7 and SRP Section 13.3. In this context, the review, using RG 1.97, addresses the performance, design, qualification, display, quality assurance, and selection of monitoring variables of radiation monitoring equipment required for accident monitoring and sampling.

The review of radiation monitoring instrumentation and controls used for occupational radiation protection or minimization of contamination, including provisions for automatic control features and interdependence with sensing elements other than radioactivity (e.g., fluid level, valve position, and system pressure, flow rate, and temperature), is performed using the guidance presented in the related DSRS sections and DSRS Section 11.6, "Guidance on Instrumentation and Control Design Features for Process and Effluent Radiological Monitoring, and Area Radiation and Airborne Radioactivity Monitoring." The review addresses the types and placement of such sensors in plant subsystems, operational ranges and qualification of sensing elements supporting the functions of radiation monitoring subsystems, the functional interdependence and the logic for alarming and controlling processes used for maintaining ORE ALARA, and minimizing facility and environmental contamination as required by 10 CFR Part 20.

14. The reviewer will evaluate the types and capabilities of radiation monitoring equipment provided to identify and assess radiological conditions near and adjacent to areas potentially impacted by AOOs and DBEs that may result in decreased quantities of consumable or variable shielding around operating or shutdown reactors containing irradiated fuel.

15. The reviewer will evaluate the radiation protection design features of secondary plant components or structures, identified in Section 12.2 as containing radioactive material resulting from direct neutron activation of secondary plant corrosion products or water that is entrained in the main steam. The evaluation should consider features provided to minimize ORE resulting from working on secondary plant components and features provided for limiting public exposure to direct radiation.

16. The reviewer will evaluate the design features provided to prevent unintended radiation exposure resulting from the operation of all cranes systems capable of moving irradiated fuel or irradiated components or capable of changing radiation zones through the movement of installed movable shielding (e.g., containment vessel bay shield blocks).

17. The reviewer will evaluate the design features provided to limit and control the radiological contaminants resulting from the direct neutron activation of the contents and chemical contaminants in the reactor building containment pool. The staff should evaluate the ability of the purification systems provided to mix and capture radioactive activation products in the pools.

18. The reviewer should evaluate the design features provided to allow maintenance, inspection, and cleaning of subsurface pool components, while reactors are operating.

19. The reviewer should evaluate the design features provided to facilitate steam generator tube inspections, including any provisions for cleaning tubes. Since steam flow is inside of the tubes instead of on the outside of the tubes as in a typical PWR, secondary plant erosion and corrosion products that typically collect on secondary side steam generator tube sheets and tube support plates may plate out on internal tube surfaces. These

deposits may adversely impact the ability to perform required steam generator tubing integrity checks.

20. The reviewer should evaluate the design features provided to ensure radiation monitoring, using 10 CFR 70.24 or 10 CFR 50.68(b), for all locations fuel where fuel is present. The staff should evaluate how the applicant meets these requirements during reactor operation, containment vessel movement, refueling, and new fuel handling.

21. The reviewer should evaluate the design features provided to minimize airborne and facility contamination from the drying of normally wetted components, such as decay heat removal exchangers.

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.

In accordance with the provisions of Sections C.I.12.3 and C.I.12.4 of RG 1.206 for DC or COL FSARs under 10 CFR Part 52 and the radiation protection aspects of 10 CFR 52.47 or 10 CFR 52.79, as well as radiation protection aspects of GDCs 19 and 61, the FSAR and amendments provide the basis for conclusions of the following type, which will be included in the staff's SER. The report will include a summary of the applicant's coverage, the staff's basis for review and acceptance criteria, and the findings of the review. The following is a brief representation of typical evaluation findings:

> The staff concludes that the radiation protection design features are acceptable and meet the relevant requirements of 10 CFR Part 20, 10 CFR Part 50, GDCs 19 and 61, and 10 CFR Part 70. This conclusion is based on the following.
>
> The radiation protection design features at [plant name] are intended to help maintain OREs within regulatory limits and ALARA, consistent with 10 CFR 20.1101(b) and the definition of ALARA in 10 CFR 20.1003, as well as RGs 8.8, and 8.10, the dose-limiting provisions of 10 CFR 20.1201, 10 CFR 20.1202, 10 CFR 20.1203, and 10 CFR 20.1204, and the non-effluent limits in 10 CFR 20.1301 and 10 CFR 20.1302. In addition, the design features are consistent with the radiation exposure and radiation source control requirements in 10 CFR 20.1406, 10 CFR 20.1601, 10 CFR 20.1602, 10 CFR 20.1801, 10 CFR 20.1901, 10 CFR 20.1902, and 10 CFR 20.1905. Many of these design features have been incorporated as a result of the applicant's radiation protection design review and from radiation exposure experience gained during the operation of other nuclear power plants. [Include examples of design features incorporated to reduce radiation to workers during maintenance operations, reduce radiation sources where operations must be performed, allow quick entry and easy access, provide remote operation capability or reduce the time required for work in radiation fields, and examples of other features that reduce radiation exposure of personnel.] These design features are consistent with those contained in RGs 8.8 and 8.38 and are acceptable.
>
> Plant design and layout facilitates the control of access to and work within plant areas in accordance with the requirements of 10 CFR 20.1601, 10 CFR 20.1602, 10 CFR 20.1901, 10 CFR 20.1902, and 10 CFR 20.1903, and access control alternatives in the NuScale Technical Specifications, and are acceptable.

Areas within the restricted area are divided into [number of zones] radiation zones. The dose rate criterion for each of these zones is derived from expected occupancy and access restrictions. These criteria are then used as the basis for the radiation shielding design. This allows for arrangements of radioactive equipment that are in accordance with the requirements of 10 CFR Part 20 and the guidelines of RG 8.8. The plant design and layout facilitates the control of access to and work within plant areas in accordance with the requirements of 10 CFR 20.1601, 10 CFR 20.1602, 10 CFR 20.1901, 10 CFR 20.1902, and 10 CFR 20.1903, and access control alternatives in the NuScale technical specifications, and are acceptable.

All plant radiation sources capable of producing radiation levels in excess of 1 Gy per hour (100 rads per hour) will be shielded and clearly marked, indicating that potentially lethal radiation fields are possible. If other-than-permanent shielding is used, administrative controls will be initiated and local audible and visible alarming monitors must be installed to alert personnel if temporary shielding is removed.

The radiation shielding will be designed to provide protection against radiation for operating personnel, both inside and outside the plant, and for the general public. The following are several of the shielding design features incorporated into [plant name]. [List several examples of shielding design features used at plant.] Some of the criteria used by [utility] in locating penetrations in shield walls at [plant name] are [list several shield penetration location criteria used]. These shielding techniques are designed to maintain personnel radiation exposures ALARA, in accordance with the provisions of RGs 8.8 and 8.10, and are acceptable.

The general shield design methodology and source term inventories used for [plant name] are similar to those from operating reactors. The basic radiation transport analysis used for the applicant's shield design is based on [list appropriate shielding computer codes used]. The applicant also used shielding information from operating nuclear plants as input data for the shield design calculations. All concrete shielding in the plant will be constructed in general compliance with RG 1.69. Where appropriate, constraints on operating conditions for shielding materials have been identified, and design features or operational programs have been identified to assure the viability of the shielding material over the life of the plant. [Where applicable, list examples of constraints on shielding material and the compensatory or monitoring elements.] The staff finds the shielding design and methodology presented in the [DC FSAR or COL FSAR] acceptable based on the DSRS criteria.

The ventilation system at [plant name] will be designed to ensure that plant personnel are not inadvertently exposed to airborne contaminants in excess of the limits provided in 10 CFR Part 20. The applicant intends to maintain personnel exposures ALARA by (1) maintaining airflow from areas of potentially low airborne concentrations to areas of higher potential concentrations, (2) ensuring negative or positive pressures to prevent exfiltration or infiltration of potential contaminants, (3) locating ventilation system intakes so as to minimize intake of potentially contaminated air from other building exhaust points, (4) providing features to prevent airborne contamination from the drying of normally wetted contaminated surfaces. These design criteria are in accordance with the guidelines of RGs 1.52 and 8.8. [List examples of exposure reduction features in the ventilation system.]

The applicant's area radiation monitoring system is designed to (1) monitor the radiation levels in areas where radiation levels could become significant and where personnel may be present, (2) alarm when the radiation levels exceed preset levels to warn of increased radiation levels, and (3) provide a continuous record of radiation levels at key locations throughout the plant. To meet these objectives, the applicant provided sufficient information describing the operational ranges and qualification of radiation detectors that support the functions of radiation monitoring subsystems, and the interdependence and the logic for alarming and actuating protective features. The applicant plans to use [number] area monitors located in areas where personnel may be present and where radiation levels could become significant. The area radiation monitoring system meets the criteria of 10 CFR 50.34(f)(2)(xvii), 10 CFR 50 Appendix E VI.2(a)(i), Item II.F.1(3) of NUREG-0737, RG 1.97 using the NuScale-specific source term, and DSRS Chapter 7 for safety-related systems, or DSRS Section 11.6, "Guidance on Instrumentation and Control Design Features for Process and Effluent Radiological Monitoring, and Area Radiation and Airborne Radioactivity Monitoring." The system is equipped with local and remote audio and visual alarms and a facility for central recording. [List examples of other area monitoring system features] The design objectives of the applicants' airborne radioactivity monitoring system are (1) to assist in maintaining occupational exposure to airborne contaminants ALARA, (2) to check on the integrity of systems containing radioactivity, and (3) to warn of unexpected release of airborne radioactivity to prevent inadvertent exposure of personnel. The applicant will install airborne radioactivity monitors in work areas where there is a potential for airborne radioactivity. These airborne radioactivity monitors have the capability to detect DAC of the most restrictive particulate and iodine radionuclides in the area or cubicle of lowest ventilation flow rate within 10 hours (usually denoted as 10 DAC-hrs). The applicant will provide portable continuous air monitors when needed to monitor air in areas not provided with fixed airborne radioactivity monitors. All airborne and area radioactivity monitors will be calibrated periodically. [List examples of other airborne radioactivity monitoring features.] The objectives and location criteria of [plant name] area and airborne radiation monitoring systems are in conformance with those portions of 10 CFR 20.1501, 10 CFR 50.34, 10 CFR 52.47, or 10 CFR 52.79, and 10 CFR 50.68 or 10 CFR 70.24, as well as RG 1.97 and RG 8.8, related to radiation and airborne radioactivity monitoring.

The objective of the applicant's accident radiation monitoring system is to provide the capability to assess the radiation hazard in areas that may be occupied during the course of an accident. The installed instruments have emergency power supplies, and the portable instruments are placed to be readily accessible to personnel responding to an emergency. The systems are designed for use in the event of an accident in terms of usable instrument range and the environment the instrument can withstand, and meet the provisions of 10 CFR 50.34(f)(2)(xvii), Item II.F.1(3) of NUREG-0737, and RG 1.97.

Instrumentation to monitor plant areas where fuel is handled and stored meets the criteria of 10 CFR 50.68 or 10 CFR 70.24, and GDC 63 in Appendix A to 10 CFR Part 50, and is acceptable.

The applicant provided a dose assessment, as described in RG 8.19, including a completed summary table of ORE estimates, sufficient detail to explain the performance of the assessment process, a systematic process for considering and evaluating dose-reducing changes in design and operations as part of the comprehensive ongoing design reviews, and a record of the review procedures, documentation requirements, and identification of principle ALARA-related changes resulting from the dose assessment, which is acceptable.

Facility design features facilitate eventual decommissioning and minimize, to the extent practicable, contamination of the facility and environment and the generation of radioactive waste in accordance with 10 CFR 20.1406.

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 SRP revision in effect 6 months before the docket date of the application. While the SRP provides generic guidance, the staff developed the SRP guidance based on the staff's experience in reviewing applications for construction permits and operating licenses for large light-water nuclear power reactors. The proposed SMR designs, however, differ significantly from large light-water nuclear power plant designs.

In view of the differences between the designs of SMRs and the designs of large light-water power reactors, the Commission issued Staff Requirements Memorandum (SRM)-COMGBJ-10-0004/COMGEA-10-0001, "Use of Risk Insights To Enhance Safety Focus of Small Modular Reactor Reviews," dated August 31, 2010. In the SRM, the Commission directed the staff to develop risk-informed licensing review plans for each of the SMR design reviews, including plans for the associated preapplication activities. Accordingly, the staff has developed the content of the DSRS as an alternative method for evaluating a NuScale-specific application submitted pursuant to 10 CFR Part 52, and the staff has determined that each application may address the DSRS in lieu of addressing the SRP, with specified exceptions. These exceptions include particular review areas in which the DSRS directs reviewers to consult the SRP and others in which the SRP is used for the review. If an applicant chooses to address the DSRS, the application should identify and describe all differences between the design features (DC and COL applications only), analytical techniques, and procedural measures proposed in an application and the guidance of the applicable DSRS section (or SRP section, as specified in the DSRS), and discuss how the proposed alternative provides an acceptable method of complying with the regulations that underlie the DSRS acceptance criteria.

The staff has accepted the content of the DSRS as an alternative method for evaluating whether an application complies with NRC regulations for NuScale SMR applications, provided that the application does not deviate significantly from the design and siting assumptions made by the NRC staff while preparing the DSRS. If the design or siting assumptions in a NuScale application deviate significantly from the design and siting assumptions the staff used in preparing the DSRS, the staff will use the more general guidance in the SRP, as specified in 10 CFR 52.17(a)(1)(xii), 10 CFR 52.47(a)(9), or 10 CFR 52.79(a)(41), depending on the type of application. Alternatively, the staff may supplement the DSRS section by adding appropriate criteria to address new design or siting assumptions.

VI. REFERENCES

1. *U.S. Code of Federal Regulations* (CFR), "Standards for Protection against Radiation," Title 10, Part 20.

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. U.S. Nuclear Regulatory Commission (NRC), "Environmental and Dynamic Effects Design Bases," General Design Criteria (GDC) 4.

5. NRC, "Reactor Coolant Pressure Boundary," GDC 14.

6. NRC, "Control Room," GDC 19.

7. NRC, "Quality of Reactor Coolant Pressure Boundary," GDC 30.

8. NRC, "Fuel Storage and Handling and Radioactivity Control," GDC 61.

9. NRC, "Monitoring Fuel and Waste Storage," GDC 63.

10. NRC, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident," Regulatory Guide (RG) 1.7 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML070290080).

11. NRC, "Measuring, Evaluating, and Reporting Radioactive Material in Liquid and Gaseous Effluents and Solid Waste," RG 1.21 (ADAMS Accession No. ML091170109).

12. NRC, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants," RG 1.52 (ADAMS Accession No. ML12159A013).

13. NRC, "Concrete Radiation Shields and Generic Shield Testing for Nuclear Power Plants," RG 1.69 (ADAMS Accession No. ML090820425).

14. NRC, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," RG 1.97.

15. NRC, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," RG 1.183 (ADAMS Accession No. ML061580448).

16. NRC, "Guide for Administrative Practices in Radiation Monitoring," RG 8.2 (ADAMS Accession No. ML110460093).

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19. NRC, “Occupational Dose Assessment in Light-Water Reactor Power Plants Design Stage Man-Rem Estimates,” RG 8.19 (ADAMS Accession No. ML003739550).

20. NRC, “Air Sampling in the Workplace,” RG 8.25 (ADAMS Accession No. ML003739616).

21. NRC, “Control of Access to High and Very High Radiation Areas of Nuclear Plants,” RG 8.38 (ADAMS Accession No. ML061350096).

22. NRC, “Clarification of TMI Action Plan Requirements,” NUREG-0737 (ADAMS Accession No. ML051400209).

23. American National Standards Institute (ANSI)/American Nuclear Society (ANS)/Health Physics Society Standards Committee (HPSSC) 6.8.1-1981, “Location and Design Criteria for Area Radiation Monitoring Systems for Light Water Nuclear Reactors.”

24. ANSI/Health Physics Society N13.1-2011, “Sampling and Monitoring Releases of Airborne Radioactive Substances from the Stacks and Ducts of Nuclear Facilities.”

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27. Camper, Larry W., NRC, memorandum to David B. Matthews and Elmo E. Collins, NRC, “List of Decommissioning Lessons Learned in Support of the Development of a Standard Review Plan for New Reactor Licensing,” October 10, 2006, (ADAMS Accession No. ML062620355).

28. NRC, “Identification and Evaluation of Facility Techniques for Decommissioning of Light Water Reactors,” NUREG/CR-3587 (ADAMS Accession No. ML081360413).

29. NRC, “Design, Inspection, and Testing Criteria for Air Filtration and Absorption Units of Normal Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants,” RG 1.140 (ADAMS Accession No. ML011710150).

30. NRC, “Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning,” RG 4.21 (ADAMS Accession No. ML080500187)

31. 10 CFR Part 70, “Domestic Licensing of Special Nuclear Material. “

32. NRC, “Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants,” RG 1.89 (ADAMS Accession No. ML003740271),

33. NRC, “Emergency Response Data System (ERDS) Implementation,” NUREG-1394, Rev. 1 (ADAMS Accession No. ML080790038).

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36. NRC, “Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants,” RG 1.143 (ADAMS Accession No. ML003740200).

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40. EPRI, “Guide for the Application and Use of Valves in Power Plant Systems,” NP-6516.

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50. NRC, “Standard Review Plans For Environmental Reviews for Nuclear Power Plants,” NUREG-1555 (ADAMS Accession No. ML13106A246).

51. NRC, “Air Sampling in the Workplace,” NUREG-1400 (ADAMS Accession No. ML13051A671).

52. NRC, “Nuclear Power Plant Instrumentation for Earthquakes,” RG 1.12 (ADAMS Accession No. ML003739944).

Appendix 12.3-12.4-A
Evaluation and Acceptance Criteria for 10 CFR 20.1406 to Support Design Certification and Combined License Applications

Purpose:

The purpose of this guidance is to clarify further the evaluation and acceptance criteria that U.S. Nuclear Regulatory Commission (NRC) staff will use in reaching a reasonable assurance finding that a DC or COL applicant has complied with the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) 20.1406, which applies to all DC and COL applications submitted after August 20, 1997. The rule requires that applicants describe how they intend to minimize, to the extent practicable, the contamination of the facility, the contamination of the environment, and the generation of radioactive waste. Applicants are also required to describe how they will facilitate decommissioning of the facility. The intent of Section 20.1406 is to emphasize to a license applicant the importance, in an early stage of planning, for facilities to be designed and operated in a way that would minimize the amount of radioactive contamination generated at the site during its operating lifetime and would minimize the generation of radioactive waste during decommissioning of the facility. Specific minimization requirements are directed toward those making an application for a new license because it is more likely that consideration of design and operational aspects that would reduce dose and minimize waste can be cost-effective at that time.

Regulatory Guide (RG) 4.21, "Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning," describes a basis acceptable to the staff for implementing the requirements of 10 CFR 20.1406. This includes a discussion of high-level objectives as well as specific actions that can be taken during design, construction, operation, and decommissioning to ensure that, to the extent practicable, contamination of the facility and the environment is minimized, radioactive waste generation is minimized, and decommissioning is facilitated.

Evaluation Criteria

If a COL application references a standard DC that meets the requirements of 10 CFR 20.1406 for design features, then the COL applicant needs to only consider those RG 4.21 criteria affecting operation and site-specific design features. At a minimum, as part of the description of design and operational features for all applicable SSCs, the applicant should also describe plans for limiting leakage, controlling the spread of contamination, detecting leaks early, allowing for appropriate and timely action to mitigate and control the spread of contamination by the future licensee, and reducing the time, effort and hazard to personnel during decommissioning activities. Where appropriate to the type of SSC being considered, the applicant should explicitly describe how these considerations are addressed in the design and operation of the SSC.

General guidance on meeting the requirements of 10 CFR 20.1406 and examples have been developed and are included as Attachment A, "Evaluation and Scoping Information for Systems, Structures and Components 10 CFR 20.1406 Design Review," to this appendix. Attachment A provides scoping information for SSCs to assist the staff in evaluation of SSCs having a potential to release radioactive materials to the facility, site, or environment that could contaminate the soil or groundwater. In addition, Attachment C cites operational experiences for various SSCs, including actual Event Notices and information included in the Liquid Radioactive Release Lessons Learned Task Force (Agencywide Documents Access and Management System (ADAMS) Accession No. ML062650312).

Regulatory Positions C.1 through C.4 in RG 4.21 are provided as specific guidance to applicants on meeting the requirements of 10 CFR 20.1406. The regulatory positions describe concepts to be implemented to provide reasonable assurance that inadvertent spills, leaks, and discharges of liquid, gaseous, and solid radioactive effluents are prevented, detected, and corrected; that the site is adequately characterized and understood; that decommissioning is planned for; and that the generation of radioactive waste is minimized. The measures to be taken by the applicant should be risk-informed and the examples described in Appendix A of RG 4.21 should be used by the applicants as guidance to determine which measures are applicable to their facility. Appendix A of RG 4.21, however, is not intended to be used as a checklist of minimally acceptable design or operational features. Alternative methods to RG 4.21 may be acceptable to meet the requirements of 10 CFR 20.1406, provided the methods are documented fully in the DC or the COL applications and accepted by the staff.

Additionally, the applicant should document that if a spill, leak, or inadvertent discharge were to occur, design or operational features would ensure that the spill, leak, or discharge would be detected promptly, and monitored and evaluated to determine the impact on the environment.

Acceptance Criteria

To determine an applicant's compliance with 10 CFR 20.1406, as it relates to describing a basis acceptable for implementing the requirements of 10 CFR 20.1406, the staff should review the applicant's description of all applicable SSCs and applicable site-specific data against the guidance contained in RG 4.21 to confirm that:

- Adequate design features exist, supplemented with operating programs, processes, and procedures (as necessary), and that these will provide reasonable assurance that spills, leaks, and inadvertent discharges of radioactive effluents will be prevented to the extent practicable, or minimized.

- In the event a spill, leak, or inadvertent discharge does occur, the staff should verify that there is reasonable assurance that it will be detected in a timely manner. For those SSCs that are typically inaccessible for routine inspection or observation, leak detection capability, to the extent practical, should allow for the identification and measurement of relatively small leak rates, depending on the concentration (e.g., several gallons per week).

- Design features should be supplemented, as necessary, by operating programs, processes, and procedures to monitor spills and leaks, and evaluate their impact to the environment.

- The site has been adequately characterized and conceptual site models have been developed that define the site hydrogeological setting, including subsurface and surface migration pathways under both pre-construction and post-construction conditions. These models are needed to assist with designing monitoring components and procedures, designing protective measures, carrying out remediation, and designing decommissioning activities.

- Design features that facilitate decommissioning (such as modular components and adequate space for equipment removal) should be described, and their role in the decommissioning process should be described. Operating procedures to minimize the amount of residual radioactivity that will require remediation at the time of decommissioning should also be described.

- The site has been designed and will be operated to minimize the generation and volume of radioactive waste, both during operation and during decommissioning.

The NRC staff's safety evaluation report (SER), related to Nuclear Energy Institute (NEI) technical report, "Generic FSAR Template Guidance for Life Cycle Minimization of Contamination," NEI 08-08A (ADAMS Accession No. ML093220530), provides the bases for the use of the template to describe an acceptable operational ground water protection program that conforms to the guidance of RG 4.21. For those licensees that elect to demonstrate compliance with the programmatic requirements of 10 CFR 20.1406 via alternate methods, SECY-04-0032, "Programmatic Information Needed for Approval of a Combined License Without Inspections, Tests, Analyses, and Acceptance Criteria," notes that in the absence of ITAAC, "fully described" should be understood to mean that the program is clearly and sufficiently described in terms of the scope and level of detail to allow a reasonable assurance finding of acceptability at the COL stage.

References:

1. *U.S. Code of Federal Regulations* (CFR), "Minimization of Contamination," Title 10 Part 20.

2. National Energy Institute (NEI) 08-08A, "Generic FSAR Template Guidance for Life Cycle Minimization of Contamination" (Agencywide Document and Management System (ADAMS) Accession No. ML093220530).

3. NRC, "Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning," RG 4.21 (ADAMS Accession No. ML080500187).

4. U.S. Nuclear Regulatory Commission (NRC), "Programmatic Information Needed for Approval of a Combined License without Inspections, Tests, Analyses, and Acceptance Criteria," SECY-04-0032, February 26, 2004 (ADAMS Accession No. ML040230079).

5. Liquid Radioactive Release Lessons Learned Task Force Final Report (ADAMS Accession No. ML062650312).

Appendix 12.3-12.4-A Attachment A

Evaluation and Scoping information for Structures, Systems, and Components 10 CFR 20.1406 Design Review

I. General Guidance

Perform an evaluation of structures, systems, and components (SSCs) that contain or could contain radioactive liquids or material. Those SSCs that have a potential to release radioactive materials to the facility, site, or environment that could contaminate the soil or groundwater should be evaluated.

The regulations require that both design and operational processes be addressed. Regulatory Guide (RG) 4.21, "Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning," describes an acceptable method for applicants to meet the regulation. The RG also includes a list of examples that may be used to determine areas to address.

Examples of SSCs, such as those listed in Attachment B of this appendix, include, but are not limited to, radioactive waste systems, building sumps and drains, spent fuel storage pools and other systems where, based on operational experience, the likelihood of such releases could occur. Typical systems and operational experience instances are included in Attachment C of this appendix.

II. General SSC Screening

If the general screening indicates review is warranted, review the final safety analysis report (FSAR) description provided to determine if the applicant has included design or operational features to address compliance with 10 CFR 20.1406. Request additional information or discuss with the Radiation Protection Branch responsible for reviewing DSRS Section 12.3-12.4, if additional information is needed.

1. Systems/Components:

   A. Does the system contain or potentially contain *radioactive materials*? (See RG 4.21, Appendix A, for examples.)

      AND;

   B. Is the system separated from the environment by a single barrier?

      - Tank/Sump with an exterior wall or floor
      - Single-walled pipe located in an area not accessible for inspection (buried pipe trench, pipe drains, etc.)

      OR;

   C. Are portions of the system located outside of a structure designed to contain a release of *radioactive materials*?

      OR;

D. Has operational experience demonstrated that the system or components has previously resulted in a release of *radioactive materials?*

2. Structures:

A. Does the structure envelope a system or components that contain or potentially contain *radioactive materials*?

AND;

B. Are there any below-grade penetrations (e.g., piping, conduit) to the environment?

OR;

C. Are there any below-grade concrete joints (e.g., floor-to-floor, walls-to-floor) that connect to the environment?

OR;

D. Does the structure contain *radioactive materials* that are separated from the environment by a single barrier? (E.g., retention pond with liner, or radioactive waste pipe running between buildings.)

Appendix 12.3-12.4-A Attachment B

Examples of Structures, Systems, and Components for 20.1406 Review

The list below provides examples of structures, systems, and components (SSCs) that typically have a potential to release radioactive material to the facility, site, or environment. Additional operating experience is provided as background information. This list is not intended to be complete and comprehensive, nor is it intended to be a checklist of minimally acceptable facility design features.

1. Spent fuel storage and transfer systems

    A. Spent fuel pool (SFP)
    B. SFP transfer canal
    C. SFP leak detection system

2. Tanks and piping

    A. Radioactive waste tanks and piping
    B. Condensate storage tank and piping
    C. High-pressure coolant injection (HPCI) and emergency service water piping
    D. Refueling water storage tank
    E. Service water and component cooling
    F. Auxiliary steam lines
    G. Cooling tower blowdown line
    H. Circulating water system piping
    I. Retention tanks
    J. Discharge canals and piping (including air relief valves on lines)

3. Drains

    A. Water treatment system drains
    B. Floor and roof drains
    C. Laundry system drains
    D. Contaminated sink drains

4. Secondary systems

    A. Plant-chilled water system
    B. Cooling tower basin

5. Radioactive waste system

    A. Waste disposal system valves
    B. Resin fill valve
    C. Retention ponds

6. Building

A. Building sumps
B. Seismic gaps
C. Joints

Appendix 12.3-12.4-A Attachment C

Operating Experiences for Review

SSC	Occurrence	Problem
Piping		
Nonsafety, HPCI suction and return piping	Underground pipe leakage	Inadequate pipe design/maintenance
Condensate tank, condensate transfer system (underground pipe)	Degraded pipe caused leak. Liquid traveled outside the protected area through an underground telephone cable conduit run.	Inadequate pipe design/maintenance
Turbine building sump discharge line	Frozen end of discharge line caused liquid to backup and leak.	No freeze protection
Radioactive waste liquid effluent release pipe	Degraded effluent line piping.	Inadequate pipe design
Coolant tower blowdown line	Cooling tower blowdown line leak because of failure in piping.	Inadequate pipe design
Turbine and waste treatment building sump discharge line	Line leaked because of degraded condition of pipe.	Inadequate pipe design
Underground pipe containing uranium bearing discharge	Pipe ruptured underground and might have been undetected for years.	Inadequate pipe design/maintenance
Sumps		
Clean sumps	Steam leaks condensed and ran into clean sumps that were routed to storm drain pond.	Inadequate maintenance
Steam Lines		
Auxiliary steam lines	Steam and liquid leaks through seals, joints, and degraded pipes.	Inadequate maintenance
Retention Ponds		
Unlined storm drain stabilization pond (SDSP)	Tritium was found in two-man holes located close to an unlined SDSP. The storm drain collector basin received overflow from the turbine building air-wash system, which contained small amounts of tritium.	Inadequate design

Appendix 12.3-12.4-A Attachment C

Operating Experiences for Review

SSC	Occurrence	Problem
Operating Practices		
Boric acid concentrator system (evaporator system) releases	Past operational practices during releases during rainy days from the system resulting in rain deposition and wash down of roof drains.	Inadequate operator procedures
Condensate transfer system	Liquid discharged from circulating water discharge tunnel through fire protection system and a portion of the service water system because of operator error.	Procedure compliance
Outdoor storage of contaminated equipment	Contamination leached from equipment onto soil.	Inadequate procedures or operational controls
Condensate storage tank	Water overflowed from condensate storage tank into a tunnel. Tunnel had potential to allow small amount of this water to permeate into the ground.	Inadequate procedures
Retention tank containing radioactive liquid	Retention tank containing uranium bearing liquid overflowed onto soil. Tank was undergoing maintenance and was not tight at the time.	Inadequate maintenance
Equipment		
Circulating water blowdown line	Vacuum breakers in blowdown line leaked while radioactive liquid traveled down the pipe.	Inadequate maintenance
Flange in feed water system venture	Leak in system. Under drain system captured most of tritium from leakage.	Inadequate design/maintenance.
Steam generator tube leak	Liquid leaked from degraded steam generator tube.	Inadequate design/maintenance
Tanks		
Storm drains around liquid waste holdup tank	Liquid leaked through cracks in the asphalt berm around a liquid waste holdup tank area into the groundwater.	Inadequate design/maintenance

Appendix 12.3-12.4-A Attachment C

Operating Experiences for Review

SSC	Occurrence	Problem
Fuel Storage and Handling		
SFP	Estimated 141,500 gallons of SFP water was released in the gap between two reactor buildings into other buildings and surrounding environment. Operational/configuration control errors resulted in deflation of SFP seals and resultant leak.	Inadequate design and operational/configuration control error.
SFP	Liner leakage and hairline crack in fuel storage building wall.	Inadequate design, bad weld
SFP	Failure of curtain drain.	Inadequate design
SFP	SFP water leaked into narrow seismic gap because of clog in tell-tale drain system.	Inadequate maintenance
SFP	Defect in liner of cask loading pool resulted in leakage from cask loading pool.	Inadequate design
SFP transfer sleeve	Leakage through fuel transfer sleeve into abandoned Unit 2 facilities.	Inadequate design