U.S. NUCLEAR REGULATORY COMMISSION

DESIGN-SPECIFIC REVIEW STANDARD FOR NuScale SMR DESIGN

11.6 GUIDANCE ON INSTRUMENTATION AND CONTROL DESIGN FEATURES FOR PROCESS AND EFFLUENT RADIOLOGICAL MONITORING, AND AREA RADIATION AND AIRBORNE RADIOACTIVITY MONITORING

REVIEW RESPONSIBILITIES

Primary - Organization responsible for the review of radiation monitoring instrumentation and sampling systems, and effectiveness of radioactive waste management systems.

Secondary - Organizations responsible for the review of instrumentation and controls, primary and secondary coolant systems, reactor and radwaste buildings ventilation systems, auxiliary cooling water systems, ventilation and air filtration systems, and control room habitability.

I. AREAS OF REVIEW

This Design-Specific Review Standard (DSRS) section complements the staff's evaluation of radiation monitoring systems that are used to monitor the presence of radioactivity in plant facilities by providing additional guidance for the review of radiation monitoring instruments and systems that do not perform safety-related functions, but are used to demonstrate compliance with the requirements for Parts 20 and 50 of Title 10 of the *Code of Federal Regulations* (CFR). The review addresses the types of and placement of radiation monitoring equipment in plant systems and facilities, operational ranges and qualification of radiation detectors in supporting the functions of monitoring subsystems, functional interdependence and logic in alarming and terminating effluent releases to unrestricted areas or diverting process or effluent streams in complying with dose limits for radiation workers and members of the public and effluent concentration limits under 10 CFR Part 20, and environmental qualification. General Design Criteria (GDC) 60 and 64 address the design basis of radiation monitoring instrumentation to control and monitor radioactive releases in the environment. GDC 4 and 10 CFR 50.49 identify the requirements for environmental qualification of equipment located in elevated radiation fields. The objective of this section is to expand the guidance on radiation monitoring instrumentation systems that do not perform safety-related functions, and to identify performance characteristics that are acceptable to the staff.

Non-safety related systems include radiological monitoring instrumentation that serves to control and monitor radiation exposures and radioactive releases, including monitoring systems that support process streams, ambient radiation exposure rates and airborne concentrations in plant areas, plant systems integrity, and building ventilation systems servicing plant areas or facilities housing irradiated fuel; demineralizers and filters; waste management equipment (as permanently installed or skid-mounted); and radioactive wastes

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and materials held in storage.

Typically, radiation monitoring systems consist of one or more remote monitors; a centrally located cabinet or console where data from radiation detectors are received, recorded, converted to meaningful radiological units, and displayed; and the necessary interconnecting cables, power supplies, actuators and motors, sampling pumps, alarms, recorders, display panels, and other auxiliary components. Some monitoring systems designs rely on the placement of a radiation detector near or within the effluent streams in order to achievethe same function without diverting any portion of the effluent stream. In specific applications, radiation monitoring systems are used to automatically initiate a protective action when exceeding a defined instrumentation alarm set-point, such as in terminating or diverting a process stream or effluent releases.

For radiation monitoring systems that supports safety-related functions, as identified by the applicant, the review of these design features and operating characteristics is performed under DSRS Chapter 7 and Standard Review Plan (SRP) Section 13.3. In this context, the review, using Regulatory Guide (RG) 1.97 and Institute of Electrical and Electronics Engineers (IEEE) Std 497-2002 and Std 603-1991 (as referenced in RG 1.97 and 10 CFR 50.55a), addresses the performance, design, qualification, display, quality assurance (QA), and selection of monitoring variables of radiation monitoring equipment required for accident monitoring and sampling. Radiological monitoring instrumentation used to initiate control room habitability functions and isolation of reactor modules is classified as a safety-related system. While the above noted technical aspects are not discussed in specific details in this DSRS Section, the review conducted here compares the design features of the proposed monitoring instrumentation and performance criteria of RG 1.97, as adopted by the applicant, and confirms that such instrumentation can provide complementary functions for systems that are used to comply with the requirements of 10 CFR Part 20 and 10 CFR Part 50. For radiation monitoring systems that support non-safety-related Three Mile Island (TMI) related functions, as identified by the applicant, the review of the design features and operating characteristics is performed using the guidance of RG 1.97 and IEEE Std 497-2002 and Std 603-1991, which address the performance, design, qualification, display, QA, and selection of monitoring variables of radiation monitoring equipment required for accident monitoring and sampling.

The review addresses (1) the process and effluent streams to be monitored by radiation detection instrumentation or sampled for separate analyses; (2) the basis for radiation monitoring areas of the plant where ambient external radiation exposure rates and airborne contamination exist; (3) the purpose of monitoring and sampling functions; (4) the parameters used to characterize, through monitoring instrumentation or sampling and analysis, radionuclide distributions and concentrations; (5) radiation monitoring in areas where special nuclear material is stored and handled; and (6) the radiation monitors used for the Emergency Response Data System (ERDS). As listed below, the areas of review are not all-inclusive and are presented as initial direction in structuring the staff's review because it is expected that there will be variations in the design and operating features of radiation monitoring equipment proposed in Integral Pressurized Water Reactor (iPWR) applications.

The areas of review include:

1. The design objectives and criteria of radiation monitoring and sampling systems, including the interfaces of skid-mounted radiation monitoring and sampling equipment

connected to their counterparts as permanently installed plant systems, as described in DSRS Section 11.5.

2. Descriptions of radiation measurement instrumentation and related sampling equipment, including redundancy and independence (where applicable); instrumentation range, calibration, and sensitivity; methods for determining alarm/trip set-points in activating alarms and terminating effluent releases or isolating processes; bases for in-plant effluent dilution when used to establish alarm set-points; types and placement of detector check sources, and diversity of components used for normal operations, anticipated operational occurrences (AOOs), and postulated accidents.

3. The locations of radiation instrumentation, monitors, and direct readouts (e.g., at a systems' local control panel and in the control room), including the proposed locations and interfaces of skid-mounted radiation monitoring equipment.

4. Layout drawings, piping and instrumentation diagrams (P&IDs), locations of process and effluent sampling points and equipment.

5. Provisions for sample collection and analysis, including sampling lines in characterizing radionuclide concentrations in process and effluent streams. This aspect of the review also addresses the bases of the locations of selected sampling stations using the U.S. Nuclear Regulatory Commission (NRC) and industry technical guidance in ensuring the collection of representative samples, while minimizing losses or deposition in sampling lines.

6. Methods used to convert raw instrumentation readings and sample analysis results into meaningful radiological results for identifying radioactivity levels or radionuclide concentrations during normal operations, AOOs, and postulated accidents in assessing radioactivity levels in releases, exposure rates, and doses in confirming compliance with NRC criteria and guidance.

7. Descriptions of the process used to evaluate instrumentation readings and results of sample analyses, and their interpretation with radiological criteria in demonstrating compliance with NRC regulations and technical specifications. If implemented, the justification for using total gross beta-gamma or alpha activity instead of conducting specific analyses, such as by radionuclide or as radioiodine and xenon dose-equivalent concentrations.

8. Measurements, analyses, or determinations made, including the bases for the interpretation of the results of sample analyses relying on the use of surrogate radionuclides as easy-to-detect in accounting for the presence of hard-to-detect radionuclides, when not specifically analyzed via the measurements of specific radionuclides.

9. Types of alarm annunciations and actions initiated by each type of instrumentation; confirmation that once tripped by an alarm set-point, the instrumentation system properly initiates and completes the expected action, such as terminating or diverting a release or process; and, if part of the design, controls for monitoring deviations of in-plant dilution and exhaust flow rates and terminating releases or isolating process flows when deviations exceed preset limits.

10. With respect to the TS on reactor coolant system (RCS) operational leakage detection instrumentation (e.g., equivalent of TS 16.3.4.15), design features of the RCS pressure boundary leakage detection systems in reliably monitoring allowable reactor coolant leakage from RCS components contained within the reactor vessel by a combination of changes in internal pressure and temperature levels, presence of steam and elevated levels of humidity, and radioactivity characterized as noble gases, fission products, and radio-iodines.

11. Design of each turbine and condenser module, including features and methods used in complying with the objectives of the TS on allowable steam generator (SG) tube leakage rates (e.g., equivalent of TS 16.3.4.13) as they relate to detection methods and sensitivities specified for radiation monitors in compliance with NRC guidance and industry standards.

12. Design provisions of equipment and measures to facilitate operation and maintenance in accordance with the guidelines of RG 1.143, as adopted by the applicant, and as referenced in topical reports, as well as previous experience with similar equipment and methods referenced in technical submittals.

13. Design features, in conformity with 10 CFR 20.1406, to reduce radioactivity levels in wastes, minimize, to the extent practicable, contamination of the facility and environment, facilitate eventual decommissioning; and minimize, to the extent practicable, the generation of radioactive waste using the guidelines of RG 1.143 and 4.21 and Nuclear Energy Institute (NEI) Template 08-08A, which was previously endorsed by the NRC (ADAMS Accession No. ML093220530).

14. Design features that allow for the return of samples collected from process and effluents streams to their origins, and prevent sampled streams from being discharged locally or released to the environment without being monitored, using the guidelines of RG 1.143 and 4.21 and NEI Template 08-08A.

15. Provisions for purging and flushing sampling lines and monitors with non-radioactive fluids (e.g., clean water, air, inert gases) and for routing purged or flushed fluids to the liquid waste management system (LWMS) and gaseous waste management system (GWMS). The review will confirm that the source of non-radioactive purging or flushing fluid is protected from backflow and radioactive cross-contamination using appropriate measures, such as check valves, backflow preventers, interlocks, differential pressures, etc.

16. Design considerations for the use of shielding (local or instrumentation mounted) commensurate with expected ambient external radiation exposure rates in maximizing the response and detection sensitivity of radiation detectors during normal operation, AOOs, and under post-accident conditions.

17. Processes used to develop, review, verify, validate, and audit digital computer software used in radiation monitoring and sampling equipment, including software used to

terminate or divert process and effluent streams. This aspect of the review addresses software developed by the applicant, purchased through a vendor, or included with the instrumentation.

18. Methods used in confirming the environmental qualification of equipment exposed to radiation over its operating life and during accident conditions under the requirements of 10 CFR 50.49. This aspect of the review addresses the radiation or radioactive environment in which equipment is located, the total expected dose during normal operation over the installed life of the equipment, and doses anticipated during accident conditions.

19. 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.

20. Description of equipment provided to comply with 10 CFR Part 50, Appendix E "Emergency Planning and Preparedness for Production and Utilization Facilities," Section VI.2(a), which requires computer systems to transmit information on plant conditions, including radioactivity release rates, unless the applicant makes a case for not including such a provision. For Pressurized Water Reactors (PWRs), the evaluation will confirm that radiation monitoring systems can transmit data on reactor coolant radioactivity, radiation levels in operating areas of the reactor and radwaste buildings, condenser air removal radiation level effluent radiation monitors, and process radiation monitor levels.

21. A comparison of the types, levels, and concentrations of radioactive material described in the design bases of the plant to methods described for specifying the type and ranges of radiation monitoring instrumentation.

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

For design certification (DC) and combined license (COL) reviews, the staff reviews the applicant's proposed ITAAC associated with 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.

<u>COL Action Items and Certification Requirements and Restrictions</u>

For a DC application, the review will also address COL action items and requirements and restrictions (e.g., interface requirements and site parameters). For a COL application referencing a DC, a COL applicant must address COL action items (referred to as COL license information in certain DCs) included in the referenced DC. Additionally, a COL applicant must address requirements and restrictions (e.g., interface requirements and site parameters) included in the referenced DC. In instances where an applicant has submitted conceptual design information for portions of the plant for which the application does not seek certification, the review should confirm that the applicant has submitted sufficient details for the staff to conduct its evaluation of the associated SSCs, assess the adequacy of interface requirements

with other SSCs that are included in the design certification, and confirm the adequacy of proposed ITAAC and methods used to verify that all interface requirements would be met by a COL applicant under the requirements of 10 CFR 52.47(a)(24) to 52.47(a)(26), 10 CFR 52.79(d)(2), and 10 CFR 52.80(a).

Operational Program Descriptions and Implementation

For COL applications that include operational programs, the review focuses on three elements. First, the review should confirm that the applicant has acknowledged the generic operational programs described in SECY-05-0197 and included them in the final safety analysis report (FSAR) as license conditions with specific implementation milestones, as noted SRP Section 13.4 and Table 13.4-x. Second, the review determines whether the applicant has identified new or complementary operational programs mandated, in part, by specific design features, and how the applicant proposes to implement these as new programs or integrate them in existing ones. Third, the review assesses whether system design features and the operating specifications of system components are adequate to achieve the objectives of each program and to demonstrate compliance with regulations using NRC and industry guidance, as applied by the applicant. When reliance on a program is used to demonstrate satisfaction of acceptance criteria and SSC performance, the specific program sub-element and implementation milestone are to be identified. If the program relies on a specific type of instrumentation and sampling systems, the review should confirm that the performance characteristics of the associated systems are commensurate with the stated objectives of the program and are acceptable for demonstrating compliance with applicable criteria.

Alternatively, a COL applicant may use industry templates and programs for the purpose of meeting the operational program milestones described in FSAR Section 13.4 and Table 13.4-x, and Section 13.5 until a plant and site-specific operational program is made available, before fuel load, for NRC evaluation. Generic Letter 89-01 and NUREG-1301 for PWR plants present further details on the format and content of the associated operational programs. DSRS Chapters 11 and 12 present specific details on the expected scope and implementation of operational programs. As applied by the applicant, the review of the implementation of operational programs will be conducted in accordance with NRC Inspection Manual Chapter (IMC) -2504, “Construction Inspection Program - Inspection of Construction and Operational Programs.” Under the directive of SECY-05-197, the implementation of operational programs does not necessitate ITAAC.

In evaluating the elements of operational programs, the review addresses:

1. The completeness and adequacy of the design features of systems and components and their operating characteristics for supporting the objectives and requirements of each operational program. In this context, systems and components include plant systems that are monitored (e.g., liquid waste management discharge line), and radiation monitoring instrumentation (e.g., a component cooling water system). The review also addresses sampling equipment (e.g., plant stack particulate and radioiodine effluents) used to characterize the radiological conditions of plant systems and effluent releases to the environment.

2. For plant-derived radionuclide distributions expected during normal operation, AOOs, and accident conditions, descriptions of instrumentation systems, radiation and radioactivity detection methods used, response characteristics, sensitivity levels and

detection limits, and detection ranges. The review addresses calibration methods and procedures in confirming that instrumentation responses meet requirements defined in operational programs or technical specifications, such as the reactor coolant system (RCS) leakage rate, steam generator tube integrity, ambient radiation exposure rates or radionuclide concentration levels, radionuclide concentrations or release rates in process and effluent streams, etc.

3. In instances when two or more radiation monitoring systems are used for accident monitoring on a single process stream or discharge point, differences in instrumentation response characteristics should be described over their expected overlapping operational ranges for noble gases, fission and activation products, and radioiodines. The information presented by the applicant should describe the basis for the response characteristics by specifying the referenced radionuclides, or radionuclide mix, forming the basis of the response and calibration. The information should identify radioactivity and concentration thresholds where sampling and analysis would be required when radiological conditions exceed the response characteristics of the instrumentation.

4. The basis for the sampling and analysis program, including sample collection frequency, types of radiological analysis, and uses of appropriate sampling media given the purpose and scope of operational programs. This aspect of the review evaluates whether the bases of related limiting conditions for operations and controls for operation are consistent with plant design features, and technical specifications, and NRC guidance given in NUREG-1301.

5. The procedures used for the prevention and detection of radioactivity in nonradioactive systems to prevent unmonitored and uncontrolled releases of radioactive material to the environment.

6. The procedures used in implementing QA and control, calibration, maintenance; use of radioactive check sources in confirming channel checks; conduct of operational functional checks; and inspection of monitoring instrumentation and sampling systems for consistency with operational and control guidelines described in NUREG-1301.

<u>Review Interfaces</u>

Systems described in the technical submittal may differ from those outlined in the DSRS or SRP. The staff should use the following recommended DSRS section interfaces as the basis for reviewing supplemental or complementary information provided in the FSAR for a specific plant design. Other DSRS sections interface with this section as follows:

1. The review of the instrumentation and sampling systems that are part of, or support, the operation of engineered safety feature (ESF) systems designed to prevent or mitigate the consequences of accidents that could result in offsite exposures comparable to the guidelines of 10 CFR Part 100 is performed under DSRS Section 7 and SRP Section 13.3, using the guidance of RG 1.97. Based on the results of the RG 1.97 evaluation, the reviewer compares the monitoring instrumentation and performance criteria and confirms whether the instrumentation can provide complementary functions for systems that are not safety-related.

2. The review of provisions for sampling during accident conditions (e.g., via the post-

accident sampling system) and in controlling sample leakage, spillage, and limiting radiation exposure to workers during sampling from process waste systems and effluent streams is conducted under DSRS Section 9.3.2, DSRS Sections 12.3-12.4, and SRP Section 13.3, using RGs 1.21, 1.97 (for the indentified accident monitoring variables), and 4.21.

3. The review of Technical Specifications (TS), as they relate to allowable RCS leakage rate, radiation instrumentation, and sampling equipment used in monitoring allowable RCS leakage rates, primary-to-secondary steam generator tube leakage rates, and steam generator tube integrity, is performed under DSRS Chapter 16, SRP Section 13.4, and SRP Section 13.5.

4. The review of design features of auxiliary systems and interfaces with radwaste management and monitoring systems is conducted under various DSRS sections, including DSRS Section 5.4.14, 9.1.3, DSRS Section 9.3.3, DSRS Section 9.3.4 and SRP Section 9.3.1, using RG 8.8, 1.52, 1.97, 1.140, 1.143, and 4.21. The systems may include the reactor pool and spent fuel pool cooling and cleanup system, condensate storage facilities, the equipment and floor drainage system, primary and secondary coolant chemistry, and ventilation systems. For the Emergency Core Cooling System (ECCS) emergency condenser and Reactor Coolant Inventory and Purification System (RCI) heat exchanger system, the review also addresses instrumentation monitoring and controlling radiological releases in the event of heat exchanger tube ruptures.

5. The organization responsible for QA performs the review of design, construction, and operation phase QA programs under SRP Chapter 17. In addition, while conducting regulatory audits in accordance with Office Instruction NRR-LIC-111 or NRO-REG-108, “Regulatory Audits,” the technical staff may identify quality-related issues. If this occurs, the technical staff should contact the organization responsible for QA to determine if an inspection should be conducted.

6. The reviews of interfaces with certified standard designs and early site permits (ESPs), COL information items, and implementation of regulatory guidance (RG, SECY, Regulatory Issue Summary (RIS), bulletins, notices, and generic letters) are performed under SRP Section 1.8 with respect to COL information items and SRP Section 1.9 on the applicant’s approach on conformance with Regulatory Criteria and guidance.

7. The review of the bases for expected radiological source terms, as they relate to process and effluent streams monitored by specific instrumentation systems, is conducted under the guidance of DSRS Sections 11.2, 11.3, 11.4, and 11.5, with the basis for the source terms provided in DSRS Sections 11.1 and 12.2 and Chapter 15.

8. The review of design features and instrumentation used for the protection of potable and sanitary water systems is conducted under DSRS Section 9.2 using RG 4.21. These systems may include potable and sanitary water systems, demineralized water makeup systems, condensate storage facilities, safety-chilled water systems, component cooling water systems, essential service water systems, turbine cooling water systems, and seal-water supply systems.

9. The review of design features and instrumentation used for the leakage detection systems, including appropriate radiation monitoring systems, is conducted under DSRS

Section 12.3-12.4 (), DSRS Section 5.2.5, and Branch Technical Position (BTP) 5-1. For design features and system operation that use NRC and endorsed industry guidance on primary and secondary coolant chemistry, the review confirms that program commitments and radiation detection methods and detector sensitivities are consistent with the information presented in NEI 97-06, underlying Electric Power Research Institute (EPRI) Guidelines, and guidance of RG 1.45, RIS 2009-02 (Rev. 1), and Information Notice 2005-24.

10. The review of design features of steam and power conversion systems and interfaces with radwaste management and monitoring systems is conducted under DSRS Section 10.4 and Section 5.4.2. The systems reviewed may include the main condenser evacuation system, the turbine gland sealing system, the condensate cleanup system, the secondary-steam system chemistry, the circulating water system, the turbine component cooling water system, auxiliary steam systems, and turbine floor drains.

11. The review of design features of plant and building ventilation systems and interfaces with radwaste management and monitoring systems is conducted under DSRS Section 9.4 using RGs 1.13, 1.52, and 1.140. The review starts with the system used to evacuate and maintain negative pressure in the reactor vessel, connections to ventilation and treatment systems, and monitoring and exhaust systems via the plant stack. Other ventilation systems reviewed include those of the reactor building, spent-fuel pool area, radwaste building and waste storage areas, localized ventilation systems servicing radiologically controlled areas, turbine building areas ventilation system and exhaust as it relates to releases from the steam jet air ejectors, and ESF atmosphere cleanup systems as it relates to plant stack releases.

12. The review of design features of areas containing new or irradiated fuel is conducted under SRP Sections 9.1.1, and 9.1.4.

13. The review of the “as low as (is) reasonably achievable” (ALARA) provisions in system design and operation to assure compliance with the occupational dose limits of 10 CFR 20.1101(b), 10 CFR 20.1201 and 10 CFR 20.1202, and Table 1 of Appendix B to 10 CFR Part 20 is conducted under DSRS Chapter 12.

14. The review of design features to reduce radioactivity levels in wastes; minimize, to the extent practicable, contamination of the facility and environment; facilitate eventual decommissioning; and minimize, to the extent practicable, the generation of radioactive waste is performed in DSRS Section 12.3 -12.4 using RGs 1.143 and 4.21, and NEI Template 08-08A to verify compliance with the requirements of 10 CFR 20.1406.

15. The review of radiation monitoring equipment subject to environmental qualification, under the requirements of 10 CFR 50.49 and GDC 4, is conducted under DSRS Section 3.11 and DSRS Chapter 12 using RG 1.97 and 1.183.

16. For COL reviews of operational programs, the review of the applicant’s implementation plan is performed under SRP Section 13.4, SRP Section 13.5.2.1, and SRP Section 13.5.2.2.

17. The review of design features of monitoring and sampling systems and components associated with the plant’s initial testing plan, description of tests, and testing

acceptance criteria is performed under DSRS Sections 9.3.2, 14.2, and 11.5 using RG 1.68.

18. The completeness of the description and design of monitoring systems and their operational features is reviewed under DSRS Section 14.3.7 and DSRS Section 14.3.8 to ensure that there is sufficient information for introduction in Tier 1, Tier 2, and Tier 2* in confirming that ITAAC are inspectable and compliance can be demonstrated with no ambiguity.

19. For portions of plant systems covered by 10 CFR Part 50, Appendix B requirements, the review of QA provisions for radiation monitoring and sampling equipment is performed using SRP Section 17.5. For portions of plant systems not covered by 10 CFR Part 50, Appendix B requirements, the review of design criteria is performed using RG 1.143 for process streams and waste treatment systems and industry guidance (American National Standard Institute (ANSI)/American Nuclear Society (ANS)-55.4-1993 (R2007), ANSI/ANS-40.37-2009, and ANSI/ANS-55.6-1993 (R2007)).

20. The review of the Maintenance Rule, using RG 1.160, as it relates to radiation monitoring instrumentation used to control and monitor effluent releases and radioactive waste management systems (RWMS) to mitigate effluent discharges during AOOs and accidents is conducted under SRP Sections 17.6 (Maintenance Rule) and 13.4 (Operational Programs), Table 13.4-x.

II. ACCEPTANCE CRITERIA

Requirements

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

1. 10 CFR 20.1201 and 10 CFR 20.1202, as they relate to the monitoring of radiation exposures to workers and associated regulatory provisions and guidance in maintaining doses as low as reasonably achievable (ALARA) under 10 CFR Part 20.1101(b). These criteria apply to exposures occurring during normal plant operations, AOOs, and accident conditions.

2. 10 CFR 20.1301 and 10 CFR 20.1302, as they relate to the monitoring of radioactivity in plant radiological effluents to unrestricted areas. These criteria apply to all effluent releases during normal plant operations, AOOs, and accident conditions.

3. 10 CFR 20.1406, as it relates to the design and operational procedures in minimizing contamination, facilitating eventual decommissioning, and minimizing the generation of radioactive waste.

4. 10 CFR 20.1501, as it relates to conducting radiation surveys and monitoring radioactivity and radiation levels in plant areas and systems, and in radiological effluents. These criteria apply to all plant activities and effluent releases resulting from operation during normal plant operations and AOOs and accident conditions.

5. Appendix B to 10 CFR Part 20, “Annual Limits on Intake (ALIs) and Derived Air

Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage," as it relates to occupational radiation exposures and radioactive effluent releases to the environment.

6. 10 CFR 50.34(b)(3), 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 during operation and the means for controlling and limiting radioactive effluents and radiation exposures within the limits set forth in part 20

7. 10 CFR 50.34a, as it relates to equipment design and procedures used to control releases of radioactive material to the environment within the numerical guidance provided in Appendix I to 10 CFR Part 50.

8. 10 CFR 50.36a, as it relates to operating procedures and equipment installed in the radioactive waste system pursuant to 10 CFR 50.34a to ensure that releases of radioactive materials to unrestricted areas are kept ALARA.

9. 10 CFR 50.49, as it relates to the environmental qualification of equipment exposed to radiation over its operating life and during accident conditions.

10. 10 CFR 50.65(a) as it relates to providing reasonable assurance that SSCs important to safety, including those that are relied upon to mitigate accidents or transients or are used in plant emergency operating procedures (EOPs) (i.e., radiation monitors described in this section and in DSRS Section 11.5, or radiation protection features described in DSRS Section 12.3) are capable of fulfilling their intended functions.

11. 10 CFR 50.68 or 10 CFR 70.24 as they relate to instrumentation for radiation monitoring in areas where special nuclear material is stored and handled.

12. 10 CFR Part 50, Appendix B, Sections XI and XII, as they relate to programs and procedures for the control of measuring and test equipment, as they apply to radiation monitoring and sampling instrumentation for systems not covered by the QA guidance of RG 1.143.

13. 10 CFR Part 50, Appendix E "Emergency Planning and Preparedness for Production and Utilization Facilities," specifically as it relates to radiation equipment provided to meet Section VI, "Emergency Response Data System," which require computer systems to transmit information on plant conditions, including radioactivity release rates, unless the applicant makes a case for not including such a provision. For PWRs, the data characterize reactor coolant radioactivity, radiation level in the reactor building, condenser air removal radiation level, effluent radiation monitor release rates, and process radiation monitor levels.

14. Appendix I to 10 CFR Part 50, as it relates to numerical guides for design objectives to meet the requirements of 10 CFR 50.34a and 10 CFR 50.36a, which specify that radioactive effluents released to unrestricted areas will be kept ALARA.

15. GDC 19, as it relates to actuation of systems that permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem (total effective dose equivalent) for the duration of the accident.

16. GDC 60, 61, 63, and 64, as they relate to monitoring and controlling effluent releases from the RWMS and designing these systems to handle radioactive materials produced during normal plant operation, including operational occurrences and postulated accidents.

17. Requirements specified in 10 CFR 50.34(f)(2)(viii), 10 CFR 50.34(f)(2)(xiv)(E), 10 CFR 50.34(f)(2)(xvii), 10 CFR 50.34(f)(2)(xxvi), and 10 CFR 50.34(f)(2)(xxvii) for monitoring gaseous effluents from all potential accident release points, and the requirements of GDC 63 and 64 in minimizing exposures to members of the public. For Part 50 requirements identified outside of 10 CFR 50.34(f)(2), the applicable provisions of 50.34(f)(2) will be evaluated in considering system operability and inter-dependency in monitoring radiological conditions during routine operation, AOOs, and accident conditions.

18. 10 CFR 52.47(b)(1), which requires that a DC application contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a facility that incorporates the design certification has been constructed and will be operated in conformity with the design certification, the provisions of the Atomic Energy Act (AEA), and the NRC's regulations.

19. 10 CFR 52.80(a), which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the COL, the provisions of the AEA, and the NRC's regulations.

20. 10 CFR 50.65(a)(1), which requires that applicants and licensees monitor the performance or condition of SSCs against licensee-established goals in a manner sufficient to provide reasonable assurance that such SSCs are capable of fulfilling their intended functions. Maintenance is important for ensuring that failure of non-safety related SSCs that could initiate or adversely affect a transient or accident is minimized, including those that are relied upon to mitigate accidents or transients or are used in plant emergency operating procedures.

DSRS Acceptance Criteria

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

1. Provisions should be made for the installation of instrumentation and monitoring equipment or sampling and analyses of all normal and potential effluent release

pathways to the environment, including nonradioactive systems that could become radioactive through interfaces with radioactive systems. For GDC 60 and 64 requirements, the system designs should comply with the applicable guidance provided in RGs 1.21, 4.15, and 1.33.

2. The gaseous and liquid process streams or effluent release points should be monitored and sampled according to DSRS Section 11.5, Tables 1 and 2. The corresponding versions of these two tables should be added as FSAR tables for radiation monitoring and sampling equipment that are specific to the design features of the proposed plant. Other process monitoring and sampling subsystems may be added to these FSAR tables beyond those listed in DSRS Section 11.5, Tables 1 and 2, if not presented elsewhere in the FSAR. With respect to the instrumentation listed in either FSAR table, the application should describe the types of sampling methods used, including continuous proportional sampling and the basis for the collection rate and frequency of periodic automatic grab sampling. For monitoring methods that use online and offline monitoring instrumentation for noble gases, radioiodines and tritium, the applicant should describe the type of continuous sampling system used, and whether the sampling system uses replaceable particulate filters (fixed or moving), radioiodine adsorber cartridges, and tritium bubblers. Generic Letter 89-01 and NUREG-1301 for PWR plants present further details monitoring and sampling requirements for instrumentation systems used to control effluent releases.

3. For systems used to monitor radiation exposure rates and airborne concentrations in plant areas or within systems, monitoring locations and sampling points should be designed in accordance with the guidance of DSRS Chapter 12. Additional monitoring and sampling subsystems may be identified in DSRS Chapter 12 beyond those listed in DSRS Chapter 12, if not presented elsewhere in the application.

4. In order to comply with GDC 63 and 64, the requirements specified in 10 CFR 50.34(f)(2)(viii), 10 CFR 50.34(f)(2)(xiv)(E), 10 CFR 50.34(f)(2)(xvii), 10 CFR 50.34(f)(2)(xxvi), and 10 CFR 50.34(f)(2)(xxvii) as they relate to radioactive waste process systems and effluent discharge paths, the design of systems and the implementation of administrative and procedural controls should conform with the guidelines of DSRS Section 11.5, Appendix 11.5-A, RG 1.21 and 4.15, and Appendix A to RG 1.33, and DSRS Section 9.3.2 and SRP Section 13.3. The design of systems should meet the provisions of NUREG-0718, NUREG-0933, and NUREG-0737, (specifically, items II.B.3 (Clarification Items 1, 3, 6, and 11), II.E.4.2 (Clarification Items 2, 5, and 7 and Attachment 1), II.F.1, Attachment 1 on noble gas effluent monitors, III.D.1.1 (Clarification Items 1 and 3), III.D.3.3 (Clarification Items 1 to 4), and the August 16, 1982, letter from D.G. Eisenhut, ADAMS Accession No. ML103420044). When two or more radiation monitoring systems are used for accident monitoring on a single discharge point, differences in instrumentation response characteristics should be described over their overlapping operational ranges for expected concentrations of noble gases, fission and activation products, and radioiodines.

5. Provisions should be made to ensure representative sampling from radioactive process streams and tank contents. Recirculation pumps for liquid waste tanks (collection or sample test tanks) should be capable of recirculating at appropriate rates to ensure thorough mixing before sampling. For gaseous and liquid process stream samples, provisions should be made for purging sampling lines and for reducing the plate-out of

radioactive materials in sample lines. Provisions for gaseous sampling from ducts and stacks should be consistent with ANSI/Health Physics Society (HPS) N13.1-2011 and ANSI N42.18-2004 on the performance of monitoring instrumentation.

6. The application should describe the process used to develop, review, verify, validate, and audit digital computer software used in radiation monitoring and sampling equipment, including software used to terminate or divert process and effluent streams. This aspect of the application addresses software developed by the applicant, purchased through a vendor, or included with the instrumentation. The applicant should provide the methods and procedures used to verify and validate computer programs, distinguish the process used to validate installed software and reprogrammable firmware, and provide the justification for using specific methods, including references to industry standards and practices.

7. The plant's Radiological Effluent Technical Specifications (RETS)/Standard Radiological Effluent Controls (SREC), and Offsite Dose Calculation Manual (ODCM) should establish set-points for actuation of automatic control features when initiating the timely actuation of isolation valves and dampers. The bases for establishing instrumentation alarm or system activation set-points should be provided. The description should address the basis for the actuation of isolation valves, dampers, etc. with automatic control features and whether they should fail in the closed or safe position. Provisions should be made to perform routine instrument calibration, maintenance, and inspections in conformance with RG 4.15 and 1.33. Instrumentation calibration procedures should consider whether the instrumentation response is expected to change given that radionuclide distributions may vary with the operating status of the plant (i.e., normal operation, AOOs, accidents, and post- accident conditions). The RETS/SREC, and ODCM should indicate the frequency of such actions. The evaluation's of the plant's proposed RETS/SREC and ODCM are addressed in DSRS Sections 11.4 and 11.5.

8. In complying with the requirements of 10 CFR Part 20.1406, the design should address those matters addressed in the guidance of RG 4.21 and industry guidance of NEI 97-06, 08-08A (ADAMS Accession No. ML093220530), and 07-07, as proposed by the industry and endorsed by the NRC. Provisions should be made to purge and drain sample streams back to the system of origin or to an appropriate waste treatment system. Purge lines should be equipped to prevent backflows and cross-contamination of non-radioactive systems supplying purging and flushing fluids. Confirm that the source of non-radioactive purging or flushing fluid is protected from backflow and radioactive cross-contamination using appropriate measures, such as check valves, backflow preventers, interlocks, differential pressures, etc.

9. Descriptions of design features and instrumentation used in primary and secondary coolant system leakage detection systems should be consistent with NEI 97-06 and underlying EPRI Guidelines, and guidance of RGs 1.11, 1.45, and 1.54, and RIS 2009-02 (Rev. 1) and Information Notice 2005-24, as they relate to detection sensitivities and detector types specified for radiation monitors to demonstrate conformance with TS for these systems.

10. Additional information on operating experience is described in various NRC and industry documents, including: Inspection and Enforcement (IE) Bulletin No. 80-10; IE Circular No. 77-14, 79-21, and 81-09; IE Information Notice (IN) No. 79-07, 79-09, 82-49, 86-42,

86-43, 91-40, 2004-05, 2006-13, and 2012-05; Regulatory Issue Summary 2008-03; ANS N42.18-2004, ANSI/ANS/HPSSC-6.8.1, ANSI/ANS-55.6-1993 (R2007), ANSI/ANS-55.4-1993 (R2007), ANSI/ANS-40.37-2009; and NUREG/CR-3587. As part of the review process, the staff should identify and point out technical and regulatory issues to applicants as they develop the design of specific systems and operational programs in ensuring that prior NRC issues identified in past IN, I&E, circulars, and RIS have been adequately considered in the FSAR.

11. The staff conducts its evaluation of conformance with technical specifications, initial plant test program, and ITAAC using the guidance of SRP Sections 13.4 and 14.3, DSRS Section 14.2 and DSRS Chapter 16, as they relate to TS, tests and test acceptance criteria, and ITAAC for radiation monitoring instrumentation and sampling systems.

12. The application descriptions of the types and ranges of radiation monitoring instrumentation should be consistent with the application's description of the kinds and quantities of radioactive materials expected in plant effluents and radioactive materials contained in wastes in meeting the requirements of 10 CFR 50.34(b)(3), 52.47(a)(5), 10 CFR 52.79(a)(3), and 10 CFR 52.157(e).

13. 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.

The description of the ERDS is acceptable if the radiation monitoring system components required by 10 CFR 50 Appendix E, Section VI.2(a) are provided with the means to transmit data. For PWRs, the evaluation will confirm that radiation monitoring systems can transmit data on reactor coolant radioactivity, radiation level in the reactor building and spent fuel pool area, condenser air removal radiation level effluent radiation monitors, effluent radiation monitor release rates, and process radiation monitor levels.

<u>Technical Rationale</u>

The technical rationale for applying these acceptance criteria to these areas of review is discussed in the following paragraphs:

1. 10 CFR 20.1201 and 10 CFR 20.1202 mandates the monitoring of radiation exposures to workers using regulatory provisions and guidance in maintaining doses ALARA under 10 CFR Part 20.1101(b). The associated criteria apply to exposures associated during normal plant operations, AOOs, and accident conditions. RG 8.8, 1.206 and 1.97, and ANSI/ANS/HPSSC-6.8.1 and ANSI/HPS N13.1-2011 contain guidance on monitoring methods and selection and placement of radiation monitoring instrumentation to control and minimize occupational radiation exposure during routine operation, accident conditions, and AOOs.

2. 10 CFR 20.1302 requires, in part, that licensees conduct surveys of radiation levels in unrestricted and controlled areas and radioactive materials in effluents released to unrestricted and controlled areas to demonstrate compliance with the radioactive dose limits contained in 10 CFR 20.1301 applicable to members of the public.

   10 CFR 20.1302 relates to methods and means in which compliance with dose limits to

individual members of the public will be achieved. This section specifies that surveys of radiation levels are conducted to demonstrate compliance with the dose limits specified in 10 CFR 20.1301. In addition, 10 CFR 20.1501 requires that radiological surveys be conducted to demonstrate compliance with the regulations of 10 CFR Part 20. These surveys use the equipment that constitutes the process and effluent radiological monitoring instrumentation and sampling systems. RG 1.21, 1.33, and 4.15, as well as industry standards (e.g., ANSI N42.18-2004 and ANSI/HPS N13.1-2011) provide additional guidance on measuring, evaluating, and reporting the results of radiological surveys.

Meeting the above requirements provides reasonable assurance that the dose limits to individual members of the public specified in 10 CFR 20.1301 and 20.1302 will not be exceeded. The review conducted in this DSRS section, with supporting information drawn from DSRS Sections 11.2, 11.3, and 11.4, evaluates the survey method and equipment used to demonstrate compliance with these regulatory requirements.

3. 10 CFR 50.34a specifies that an application to construct or operate a nuclear power plant describe the design of equipment installed to maintain control of radioactive materials in plant effluents produced during normal operation, including expected operational occurrences.

   10 CFR 50.34a relates to DSRS Section 11.6 because processes to monitor and survey radioactive materials in liquid and gaseous effluent streams released to the environment provide crucial information for establishing controls over these effluents. 10 CFR Part 50.34a requires that descriptions of the equipment and procedures used to control gaseous and liquid eflfuents be included for all systems that are used to process radioactive wastes. As described in this DSRS section, 10 CFR 50.34a mandates a description of the equipment used to monitor and survey effluents. RG 1.143 (Regulatory Positions C.4 and C.7) provides additional guidance for the design, construction, installation, and testing of radioactive waste management SSCs. Conformance with the guidance is addressed separately in DSRS Section 11.2 for the LWMS, Section 11.3 for the GWMS, and Section 11.4 for the solid waste management system (SWMS).

   Meeting the requirements of 10 CFR 50.34a provides reasonable assurance that the level of radiation in effluents from nuclear power plants will meet the ALARA criterion and dose objectives of Appendix I to 10 CFR Part 50. The review conducted in this DSRS section, with supporting information from DSRS Sections 11.2, 11.3, and 11.4, evaluates the methods used to demonstrated compliance with these regulatory requirements.

4. 10 CFR 50.36a specifies, in part, that licenses for nuclear power reactors will include TS requiring that operating procedures be developed for the equipment specified in this regulation.

   In accordance with 10 CFR 50.36a, licensees must include TS (RETS/SREC) as part of the operating procedures related to radiological monitoring and sampling equipment and as part of the requirements for administrative controls and surveillance. The ODCM consolidates the plant's TS and related radiological effluent controls, as stated in Generic Letter 89-01, NUREG-1301, NUREG-0133, and NUREG-0543.

Meeting the requirements of 10 CFR 50.36a provides reasonable assurance that the levels of radioactivity in effluents from nuclear power plants will meet the ALARA criterion and result in doses to members of the public that are a small fraction of the 10 CFR 20.1301 limits. The review conducted under this DSRS section, with supporting information derived from the review conducted under DSRS Sections 11.2, 11.3, 11.4, and 11.5 evaluates the methods used to demonstrate compliance with these regulatory requirements.

5. Appendix I to 10 CFR Part 50 provides numerical guides for the ALARA criterion for radioactive materials released by light-water-cooled nuclear power reactors.

   10 CFR 50.34a and 10 CFR 50.36a contain provisions designed to ensure that releases of radioactive materials, as liquid and gaseous effluents, from nuclear power reactors to unrestricted areas during normal operation, including expected operational occurrences, are ALARA. Part 50, Appendix I provides specific numerical criteria and guidance for meeting this requirement.

   Meeting the requirements of the ALARA criterion provides reasonable assurance that offsite doses to any individual from normal operations and from AOOs will not result in exposures in excess of the numerical guides specified in Section II of Appendix I to 10 CFR Part 50. The review conducted under this DSRS section, with supporting information derived from the review conducted under DSRS Sections 11.2, 11.3, 11.4, and 11.5 evaluates the methods used to demonstrate compliance with these regulatory requirements.

6. Compliance with GDC 60 requires that the nuclear power plant design include mechanisms to control the release of radioactive materials in gaseous and liquid effluents and handle radioactive solid wastes produced during normal reactor operation, including AOOs.

   GDC 60 applies to DSRS Section 11.6 because mechanisms to control releases of radioactive effluents should include, among other components, equipment and operating procedures to provide monitoring, sampling, and surveillance of effluent streams that may contain radioactive materials. RG 1.143 provides guidance on the design, construction, installation, and testing of radioactive waste management SSCs.

   Meeting the requirements of GDC 60 provides reasonable assurance that releases of radioactive materials during normal operations and AOOs will not result in offsite radiation doses that exceed the limits and design objectives specified in the regulations. The review conducted under this DSRS section, with supporting information derived from the review conducted under DSRS Sections 11.2, 11.3, 11.4, and 11.5, evaluates the methods used to demonstrate compliance with the regulatory requirements.

7. Compliance with GDC 63 and 64 addresses the means to (1) monitor for excessive radiation levels in radioactive waste facilities, and (2) monitor radioactivity in effluent discharge paths and the plant's environs during normal operation, AOOs, and postulated accidents.

   GDC 63 and 64 relate directly to DSRS Section 11.6 because they focus on monitoring radiation levels within the plant, as well as radioactivity levels in effluent streams and

plant environs, during normal operations, AOOs, and postulated accidents. The requirements specified in 10 CFR 50.34(f)(2)(viii), 10 CFR 50.34(f)(2)(xiv)(E), 10 CFR 50.34(f)(2)(xvii), 10 CFR 50.34(f)(2)(xxvi), and 10 CFR 50.34(f)(2)(xxvii) are consistent with the requirements of GDC 64. RG 1.21, 1.33, and 4.15 provide guidance on radiological monitoring programs for normal operation and AOOs, while ANSI N42.18-2004 and ANSI/HPS N13.1-2011 provide guidance on the selection and use of continuous radiation monitoring equipment and methods in sampling airborne radioactive materials in nuclear facilities. The guidance cited above also address requirements for QA programs.

In addition to RG 4.15, the review addresses the process used to develop, review, verify, validate, and audit digital computer software used in radiation monitoring and sampling equipment, including software used to terminate or divert process and effluent streams. This aspect addresses software developed by the applicant, purchased separately through a vendor, or included with the instrumentation.

Meeting the requirements of GDC 63 and 64 provides reasonable assurance that the levels of radioactivity in effluents from nuclear power plants will not exceed specified limits and design objectives. The review conducted under this DSRS section, with supporting technical information obtained from the review conducted under DSRS Sections 11.2, 11.3, and 11.4, evaluates the method used to demonstrate compliance with these regulatory requirements.

8. The requirements under 10 CFR 50.34(f)(2)(viii), 10 CFR 50.34(f)(2)(xiv)(E), 10 CFR 50.34(f)(2)(xvii), 10 CFR 50.34(f)(2)(xxvi), and 10 CFR 50.34(f)(2)(xxvii) specify that systems be provided to monitor gaseous effluents from all potential accident release points.

   When examining the applicant's system for sampling process streams and effluents under accident conditions, the reviewer considers RG 1.97 and 1.101; NUREG-0933, NUREG-0737 (specifically, items II.B.3 (Clarification Items 1, 3, 6, and 11), II.E.4.2 (Clarification Items 2, 5, and 7 and Attachment 1), II.F.1, Attachment 1 on noble gas effluent monitors, III.D.1.1 (Clarification Items 1 and 3), III.D.3.3 (Clarification Items 1 to 4), and the August 16, 1982, letter from D.G. Eisenhut); DSRS Section 9.3.2; and the applicant's Emergency Plan and implementation procedures, as described in SRP Section 13.3. Based on the result of the evaluation guided by RG 1.97, the reviewer compares the proposed design specifications of radiation monitoring instrumentation and performance criteria against RG 1.97 and confirms that the instrumentation can provide complementary functions for systems that are used to comply with 10 CFR Part 20 and 10 CFR Part 50. The principal functional capabilities should include:

   A. Purging lines to flush sampling lines and return purging fluids to the appropriate subsystems for collection or treatment.

   B. Design features minimizing sample loss by reducing the length and bending radii of sample lines to ensure that samples are representative of reactor primary coolant, reactor steam, secondary coolant, and process steam.

   C. Maximize sample quality and integrity by reducing conditions that would distort a sample's chemical and physical compositions.

D. Preventing blockage of sample lines by foreign matter or dual phase flows.

E. Minimizing sampling flow restrictions and using remotely operated isolation valves to limit reactor coolant loss from ruptured sample lines.

F. Using shortest possible sample lines to minimize the volume of process fluid taken from each reactor module, system processes, and effluent streams.

G. If inline radiation monitoring is used, the design should provide backup provisions for grab sampling under an expected range of radiological conditions and provide the means to protect personnel from excessive radiation levels and airborne concentrations at sampling stations.

H. Verifying that specifically identified radioactive release points have provisions for automatic termination or diversion of releases when exceeding predetermined alarm levels.

I. Confirming that the design allows detectors to be replaced or decontaminated without opening the boundary of the process system or without losing the capability to isolate the system or divert effluents to tanks or standby treatment systems (as appropriate).

J. When two or more radiation monitoring systems are used for accident monitoring on a single discharge point or process stream, confirm that differences in instrumentation response characteristics, given the design features of the proposed instrumentation systems,are described over their stated overlapping operational ranges for noble gases, fission and activation products, and radioiodines using the guidance of RG 1.97.

K. Confirming that proposed instrumentation calibration methods consider whether instrumentation responses are expected to change given that radionuclide distributions may vary with the operating status of the plant (i.e., normal operation, AOOs, and during accident and post-accident conditions).

9. 10 CFR 50.49 and GDC 4 require that systems important to safety be capable of performing their design functions when exposed to external radiation during normal operating conditions, AOOs, and accident conditions.

   In the context of this review, the evaluation should consider ambient radiation levels from all sources of radioactivity, in which instrumentation operates, and include the total dose expected during normal operation over the installed life of the equipment, and total dose anticipated during accident conditions. The review should confirm that the applicant has identified all appropriate systems and components and provided information describing methods used in qualifying the equipment located in plant areas with elevated radiation levels. For plants using Revision 4 of RG 1.97, accident monitoring equipment identified as Type A, B, or C variables in accordance with that guide should be environmentally qualified as required by 10 CFR 50.49. Type D variables should be environmentally qualified for the particular accident's postulated environment at the installed location in accordance with the plant's licensing basis.

10. 10 CFR 50.65(a) requires the implementation of a program to monitor the performance or operating conditions of SSCs important to safety, including systems and components that are relied upon to mitigate, transients or accidents.

    When examining the applicant's monitoring program, the reviewer applies the guidance of RG 1.160 as it relates to the maintenance of radiation monitoring and sampling systems that are important to the protection of public health and safety. The evaluation of the maintenance program should consider subsystems that include radiation monitoring equipment, as described in this DSRS section, radiation protection features described in DSRS Section 12.3-12.4, and those that are used in plant emergency operating procedures. The review should confirm that the radiation monitoring equipment and their use, in the context of emergency response, provide the means to assess ambient radiation levels and concentrations of radioactive materials in process streams and effluents and can be relied upon to mitigate AOOs, plant transients, or accident conditions.

11. 10 CFR 20.1301(e) requires nuclear power plants to comply with the Environmental Protection Agency's (EPA) generally applicable environmental radiation standards of 40 CFR Part 190 for operations which are part of a nuclear fuel cycle. The standards specify dose limits for members of the public.

    The review for compliance with the EPA's generally applicable environmental radiation standards of 40 CFR Part 190, as implemented under 10 CFR 20.1301(e), is conducted under DSRS Sections 11.2 to 11.5 for source terms and doses due to liquid and gaseous effluents associated with the operation of the LWMS, GWMS, and SWMS. Doses associated with external radiation from buildings and sources of radioactivity contained in systems and components are evaluated under DSRS Section 12.3-12.4. The review focuses on methods used to assess the total dose from sources of radioactivity, external radiation exposures from waste processing buildings, waste storage buildings, waste storage tanks, and temporary waste storage or staging areas, and spent-fuel storage.

12. 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 mandates that appropriate systems be provided to adequately monitor and initiate safety actions in facility handling areas and 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.

13. 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.

14. Compliance with the requirements of 10 CFR Part 50, Appendix E, Section VI.2(a), ensures the provision of accurate and timely radiation monitor data needed to determine core and coolant system conditions well enough to assess the extent or likelihood of

core damage and to determine conditions within each reactor module and reactor building well enough to assess the likelihood and consequence of system failure.

15. 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

These review procedures are based on the identified DSRS acceptance criteria. For deviations from these acceptance criteria, the staff should review the applicant's evaluation of how the proposed alternatives provide an acceptable method of complying with the relevant NRC requirements identified in Subsection II. The review should confirm that the applicant has submitted sufficient information for the staff to conduct an independent evaluation of any proposed alternative method and demonstration of compliance with NRC regulations and DSRS acceptance criteria and supporting regulatory guidance.

1. Selected Programs and Guidance - In accordance with the guidance in NUREG-0800, "Introduction - Part 2: Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: Integral Pressurized Water Reactor Edition" (NUREG-0800 Intro Part 2) as applied to this DSRS Section, the staff will review the information proposed by the applicant to evaluate whether it meets the acceptance criteria described in Subsection II of this DSRS. As noted in NUREG-0800 Intro Part 2, the NRC requirements that must be met by an SSC do not change under the SMR framework. Using the graded approach described in NUREG-0800 Intro Part 2, the NRC staff may determine that, for certain structures, systems, and components (SSCs), the applicant's basis for compliance with other selected NRC requirements may help demonstrate satisfaction of the applicable acceptance criteria for that SSC in lieu of detailed independent analyses. The design-basis capabilities of specific SSCs would be verified where applicable as part of completion of the applicable ITAAC. The use of the selected programs to augment or replace traditional review procedures is described in Figure 1 of NUREG-0800, Introduction - Part 2. Examples of such programs that may be relevant to the graded approach for these SSCs include:

    - 10 CFR Part 50, Appendix A, General Design Criteria (GDC), Overall Requirements, Criteria 1 through 5
    - 10 CFR Part 50, Appendix B, Quality Assurance (QA) Program
    - 10 CFR 50.49, Environmental Qualification of Electrical Equipment (EQ) Program
    - 10 CFR 50.55a, Code Design, Inservice Inspection and Inservice Testing (ISI/IST) Programs
    - 10 CFR 50.65, Maintenance Rule requirements
    - Reliability Assurance Program (RAP)
    - 10 CFR 50.36, Technical Specifications
    - Availability Controls for SSCs Subject to Regulatory Treatment of Non-Safety Systems (RTNSS)
    - Initial Test Program (ITP)

- Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC)

This list of examples is not intended to be all-inclusive. It is the responsibility of the technical reviewers to determine whether the information in the application, including the degree to which the applicant seeks to rely on such selected programs and guidance, demonstrates that all acceptance criteria have been met to support the safety finding for a particular SSC.

2. In accordance with 10 CFR 52.47(a)(8),(21), and (22), and 10 CFR 52.79(a)(17), (20) and (37), for design certification or combined license applications submitted under Part 52, the applicant is required to (1) address the proposed technical resolution of unresolved safety issues and medium- and high-priority generic safety issues which are identified in the version of NUREG-0933 current on the date up to 6 months before the docket date of the application and which are technically relevant to the design; (2) demonstrate how the operating experience insights have been incorporated into the plant design; and, (3) provide information necessary to demonstrate compliance with any technically relevant portions of the Three Mile Island requirements set forth in 10 CFR 50.34(f), except paragraphs (f)(1)(xii), (f)(2)(ix), and (f)(3)(v) for a DC application, and except paragraphs (f)(1)(xii), (f)(2)(ix), (f)(2)(xxv), and (f)(3)(v) for a COL application. These cross-cutting review areas should be addressed by the reviewer for each technical subsection and relevant conclusions documented in the corresponding safety evaluation report (SER) section.

The types of systems of concern to this DSRS section are non-safety related systems. Non-safety related systems include radiological monitoring instrumentation that serves to control radiological releases to the public; supports process streams; indicates ambient radiation exposure rates and airborne concentrations in plant areas; indicates plant system integrity; monitors and controls building ventilation systems servicing plant areas or facilities housing new or irradiated fuel, demineralizers and filters, or waste management equipment (permanently installed or skid-mounted); and monitors radioactive wastes and materials held in storage.

The information describing the design features of radiation monitoring instrumentation and sampling systems are provided in the DC application, the update of the final safety analysis report, or the COL application to the extent not addressed in a referenced certified design. In instances where an applicant has submitted conceptual design information for portions of the plant for which the application does not seek certification, the review should confirm that the applicant has submitted sufficient details for the staff to conduct its evaluation of the associated SSCs, assess the adequacy of interface requirements with other SSCs that are included in the design certification, and confirm the adequacy of proposed ITAAC and methods used in verifying that all interface requirements have been met by a COL applicant under the requirements of 10 CFR 52.47(a)(24)–(26), 10 CFR 52.79(d)(2), and 10 CFR 52.80(a).

The method for providing source checks of radiation monitoring instruments should be reviewed. Source checks should verify that the detector and the electronics associated with the detector are functioning properly. Light emitting diodes (LED) or electronic signal injection (e.g., a steady-state current or a specified pulse rate) type source checks that check only the electronics may not be sufficient to ensure the full functional capability of the instrumentation, which includes the radiation detector and its supporting electronics, (e.g., a base pre-amplifier).

In Plant Radiation and Radioactivity Monitoring for Radiation Protection

With respect to radiation protection of plant workers, the description, operational characteristics, and placement of fixed radiation monitoring systems to monitor ambient radiation levels and airborne radioactivity are dictated by radiation and radioactivity levels anticipated during normal operations, AOOs, and accidents. The criteria for the placement of radiation monitoring equipment and methods for obtaining representative air samples in work zones and ventilation system ductwork, in compliance with regulatory requirements (doses and internal exposures) and guidance, are presented in DSRS Sections 12.3-12.4 and 12.5 and are not repeated here. The regulatory requirements address dose limits to plant workers, occupational concentration limits for inhalation and ingestion, and ALARA provisions in minimizing radiation exposure within the context of an operational program relying on installed radiation monitoring and sampling equipment with specific operating characteristics.

RG 8.8 and 1.206, and standards ANSI/ANS/HPSSC-6.8.1 and ANSI N13.1-2011 contain guidance for the selection and placement of radiation monitoring instrumentation for monitoring areas where reactor fuel is stored or handled. The proposed selection and placement of radiation monitoring instrumentation in areas where fuel is stored will be deemed acceptable if they meet the criteria of 10 CFR 50.68 or 10 CFR 70.24, provisions of 10 CFR 20.1501.

Radiation Monitoring of Process Streams and Effluents

For radioactive liquid and gaseous effluents, the process and effluent radiological monitoring instrumentation and sampling systems (PERMISS) are used to monitor process streams and effluents from the LWMS, GWMS, and SWMS. The PERMISS includes subsystems used to collect process and effluent samples during normal operation, AOOs, and under accident and conditions. Compliance with regulatory requirements (dose and effluent concentration limits) and use of applicable NRC and industry guidance are presented in DSRS Sections 11.2 to 11.5 and are not repeated here. The regulatory requirements address dose limits to members of the public, concentration limits for liquid and gaseous effluent discharges, and design objectives and ALARA provisions in controlling and monitoring effluent discharges in plant environs within the context of established operational programs relying on installed radiation monitoring and sampling equipment with specific operating characteristics.

With respect to system design features, the review addresses surveillance requirements and controls; operational conditions of radiation monitoring and sampling equipment; required number of operational channels; conduct and frequencies of channel checks, source checks, channel calibrations, and channel functional checks; compliance with action statements and remediation whenever the number of operational channels and applicability are less than the required minimum; sampling and analysis programs for continuous and batch mode releases, including provisions for the collection of grab and composite samples; and derivations of the lower limit of detections by categories of effluents or radionuclides and types of radiological analyses. For designs that rely on skid-mounted radiation monitoring and sampling systems connected to permanently installed process and waste management systems, the review should determine whether the applicant has provided sufficient plant-specific information describing how the design and operating features are integrated with the PERMISS. The review should address the requirements of 10 CFR Part 20.1406 and conform with the

guidance of RG 4.21 and 1.143, IE Bulletin 80-10, ANSI/HPS-13.1-2011 and ANSI N42.18-2004, and NEI 08-08A.

The review should confirm that the applicant has (1) identified all liquid and gaseous effluent release points and the types and locations of installed radiological instrumentation used to monitor and control effluent releases, (2) described parameters and provide justification of values used to derive effluent release rates and alarm set-points, including the bases of in-plant dilution factors for liquid effluents, (3) provided specifications for maximum radioactivity levels in tanks containing liquids and descriptions of protective measures applied to spills and leaks from such tanks, (4) described criteria used to determine the operability of waste treatment systems and requirements in conducting dose projections, such as whenever treatment systems are not fully utilized, or in assessing monthly, quarterly, and yearly doses, and (5) defined administrative and operational procedures associated with the implementation of the ODCM.

The programs identified in the administrative controls section of the TS and elements of the RETS/SREC are reviewed using the provisions of Generic Letter 89-01 and NUREG-1301. The programs include the RETS or SREC as they relate to the plant's ODCM. The review includes the evaluation or development of appropriate limiting conditions for operation or controls and their bases, consistent with plant design features and identified release points. The reviewer determines that the elements and scope of the programs identified in the administrative controls section of the TS agree with the requirements identified as a result of the staff's review. The evaluation of the above noted programs is conducted as part of the review of DSRS Sections 11.4, 11.5, and 13.4 and is not repeated here.

<u>Radiation Monitoring for Non-Effluent Process Streams</u>

For liquid and gaseous process radiation monitoring equipment not covered by the ODCM, the review should confirm that the applicant provides information describing methods and procedures that will be used in deriving lower limits of detection or detection sensitivities, and set-points (alarms and process termination/diversion) for process radiation monitoring equipment. Similarly, the applicant is responsible for developing a plant-specific process and radiological sampling and analysis plan for such systems, including provisions describing sampling and analytical frequencies, and radiological analyses for the expected types of liquid and gaseous samples and waste media generated by such plant systems. The review should confirm that the proposed sampling and analytical frequencies, and radiological analyses are commensurate with expected levels of radioactivity in associated systems. The review should also confirm that procedurally, the results of such radiological analyses would be used to assess the performance of process and effluent treatment systems.

The description, operational characteristics, and placement of radiation monitoring systems to monitor radiation levels and radioactivity levels in primary and secondary systems are dictated by operating specifications for confirming the integrity of the reactor coolant system and allowable leak rates, and in maintaining primary and secondary coolant chemistry. Among others, such systems include the reactor coolant pressure boundary leakage detection system, the component cooling water system, and the spent fuel pool cooling and cleanup system. Depending on the design features of these systems, the staff may consider in its review associated system interfaces and operational considerations.

The review of design features and instrumentation used for the leakage detection systems,

including appropriate radiation monitoring systems, complements the reviews conducted under DSRS Sections 12.3-12.4 and 5.2.5 and BTP 5-1, as specified for radiation monitors used in conformance with NEI and EPRI guidelines, Technical Specifications Task Force recommendations, and guidance of RGs 1.45 and 1.54, and RIS 2009-02 (Rev. 1) and Information Notice 2005-24, as referenced herein. To the extent not covered in DSRS Sections 5.2.5 and 12.3-12.4, and SRP Section 10.4.8, the review will address the placement of radiation detectors and sampling lines, types of detectors and detection sensitivities, selection of radionuclides forming the basis of instrumentation responses, assumed primary coolant concentrations in deriving detection sensitivity thresholds in complying with technical specifications, and placement of radiation detectors in plant locations with low ambient external radiation levels to maximize instrumentation response times.

With respect to the TS on RCS operational leakage detection instrumentation (e.g., equivalent of TS 16.3.4.15), the review will evaluate the capability of the evacuation system to maintain negative pressure within each reactor vessel containment, whether the evacuation system will be operating in a continuous or intermittent mode, connection of each evacuation systems to an appropriate treatment system (gaseous and liquid for condensates), and radiation monitoring prior to directing the exhaust to the plant stack. The review should determine whether the RCS pressure boundary leakage detection systems can reliably monitor reactor coolant leakage from RCS components and contained within the reactor vessel by a combination of changes in internal pressure and temperature levels, presence of steam and elevated levels of humidity and radioactivity as noble gases, fission products, and radio-iodines, and plate-out of particulate radionuclides on internal surfaces of the containment vessel. To the extent not covered in DSRS Sections 5.2.5, 11.2, 11.3, 11.5, and 12.3-12.4, the review should address the placement of radiation detectors and sampling lines, types of detectors and detection sensitivities, selection of radionuclides forming the basis of instrumentation responses, assumed primary coolant concentrations in deriving detection sensitivity thresholds in complying with technical specifications, and placement of radiation detectors in plant locations with low ambient external radiation levels to maximize instrumentation response times.

Given that the design does not include a steam generator (SG) blowdown and treatment system, the review will consider the capability of the condensate polishing demineralizers to collect, direct, and monitor concentrations of fission products (e.g., radio-iodines, noble gases, and other non-condensable gases) from the secondary side in the event of SG tube failures. The review will evaluate design features to collect exhausts from the main condenser evacuation system (MCES) and turbine gland sealing system (TGSS) for processing and monitoring. The review will consider necessary interfaces with liquid and gaseous waste management systems, decay heat removal system heat exchangers, feedwater condensate storage tanks, and radiation monitoring systems in avoiding uncontrolled and unmonitored releases to the environment and cross-contamination of non-radioactive systems. NRC requirements and NRC and industry guidance are provided in Part 10 CFR 20.1406, DSRS Sections 5.4.2, 10.4.8, 11.2, 11.3, 11.4, and 12.3-12.4, and Branch Technical Position (BTP) 5-1 (Monitoring of Secondary Side Water Chemistry in Pressurized Water Reactor (PWR) Steam Generators), RG 4.21, and NEI 08-08A.

The review of the design of each turbine and condenser module should determine whether the design features are based on existing NRC and industry guidance or rely on alternate methods in complying with the objectives of the TS on SG tube leakage rates (e.g., equivalent of TS 16.3.4.13). The review should consider current NRC and industry guidance,

as described in DSRS Sections 5.4.2, 10.4.8, 11.2, 11.3, 11.5, and 12.3-12.4, and Branch Technical Position (BTP) 5-1 (Monitoring of Secondary Side Water Chemistry in Pressurized Water Reactor (PWR) Steam Generators), as they relate to detection sensitivities and detector types specified for radiation monitors provided for compliance with NEI 97-06, underlying EPRI Guidelines, and guidance of RGs 1.29 and 1.45, and RIS 2009-02 (Revision 1) and IN 2005-24.

The evaluation should review the descriptions of procedures used to implement QA and control, calibration, maintenance, use of radioactive check sources to confirm operation of instrument channels, conduct of operational functional checks, and inspection of monitoring instrumentation and sampling systems.

The review of design features of steam and power conversion systems and interfaces with radwaste management and monitoring systems supports the review conducted under DSRS Section 10.4 and Section 5.4.2 using RG 1.143 guidance in assigning safety classifications. The systems may include the main condenser evacuation system, turbine gland sealing system, condensate cleanup system, secondary-steam system chemistry, circulating water system, turbine component cooling water system, and turbine floor drains. The staff's review addresses the placement of radiation detectors and sampling lines, types of detectors and detection sensitivities, selection of radionuclides forming the basis of instrumentation responses, assumed coolant (primary and secondary) and steam concentrations in deriving detection sensitivity thresholds to comply with technical specifications, and placement of radiation detectors in plant locations with low ambient external radiation levels in maximizing instrumentation response characteristics.

Radiation Monitoring System Design Features Credited for Compliance with 10 CFR 20.1406

The review considers information describing design features that will minimize, to the extent practicable, contamination of the facility and environment; facilitate eventual decommissioning; and minimize, to the extent practicable, the generation of extraneous radioactive wastes associated with the operation of the system as a result of operator errors and processing equipment failures or malfunctions. The radiation instrumentation and sampling systems are reviewed to ensure that the design includes provisions to prevent and collect leakage and spillage of radioactive materials associated with their operation, including sample collection, processing, storage, and operation of skid-mounted monitoring and sampling equipment that conform to the guidelines of RG 1.143, 4.21 and 8.8, IE Bulletin No. 80-10 to demonstrate compliance with the requirements of 10 CFR 20.1406. Also, the review should consider the information contained in the update to the FSAR or the COL application, to the extent not addressed in a referenced certified design.

The review should confirm whether design features are included to return samples collected from process and effluents streams to their origins, and to prevent sampled streams from being discharged locally or released to the environment without being treated and monitored using the guidelines of RG 1.143, 4.21 and 8.8, IE Bulletin No. 80-10, and NEI Template 08-08A. The review should evaluate provisions for purging and flushing sampling lines and monitors with non-radioactive fluids (e.g., clean water, air, inert gases) and for routing purged or flushed fluids to the appropriate RWMS process stream. In addition, the review should confirm that the source of non-radioactive purging or flushing fluid is protected from radioactive cross-contamination using appropriate measures, such as check valves, backflow preventers, interlocks, differential pressures, etc.

The evaluation should review the descriptions of procedures used in implementing QA and quality control, calibration, maintenance, use of radioactive check sources in confirming operation of instrument channels, conduct of operational functional checks, and inspection of monitoring instrumentation and sampling systems.

Radiation Monitoring Systems with Automatic Control Features

In specific applications, radiation monitoring systems are used to automatically initiate a protective action when radiation or radioactivity levels exceed instrumentation alarm set-point, such as terminating or diverting a process stream or effluent releases. The initiation of control or protective actions may rely solely on the presence of radioactivity or be linked to the status other plant system parameters as functional interdependence and logic in alarming and terminating or diverting process or effluent streams. The alarm set-point may represent radioactivity concentration levels and release rates, or signals other than radioactivity (e.g., fluid level, valve/damper position, system pressure, flow rate, and temperature. Other considerations may include determining whether system logic demands that a valve or damper should fail in the closed position in protecting the system from further contamination, terminating releases to the environment, or diverting process streams or effluents to appropriate treatment subsystems.

The review addresses the types and placement of such sensors in plant subsystems, operational ranges and qualification of sensing elements in supporting the functions of radiation monitoring subsystems, and functional interdependence and logic in alarming and terminating or diverting process or effluent streams. Common criteria used in establishing and setting alarm set-points are: (1) complying with dose limits for plant workers and members of the public under 10 Part 20 (Subparts B, C, and D), (2) complying with effluent concentration limits under 10 CFR Part 20 (Appendix B, Table 2 effluent concentration limit), (3) complying with the design objectives and ALARA provisions of 10 CFR Part 50, Appendix I, Section II, and (4) in protecting the integrity of plant systems, such as protecting ion-exchange beds from excessive temperatures, preventing system overflows which could lead to spills and leaks, avoiding the cross-contamination of otherwise non-radioactive systems, etc. Depending on specific design features and types of automatic control features, the staff's evaluation initiated under this DSRS section may involve complementary reviews of other systems in considering component interfaces, such as instrumentation and controls, electrical power distribution to valve and damper actuators, provisions in flushing and purging sampling lines, and system engineering (e.g., balance of plant systems and ventilation systems servicing ambient areas of the plant containing radioactive materials and collecting offgases from system and component vents).

The evaluation should review the descriptions of procedures that will be used to operate the radiation monitoring systems. The evaluation should address instrumentation calibration, maintenance, use of radioactive check sources in confirming operation of instrument channels, sample extraction, and conduct of operational functional checks. Separately, the evaluation should consider the implementation of the quality assurance program and routine inspections of the radiation monitoring systems.

Radiation Monitoring Systems Used for AOOs and Accident Conditions

While the review of safety-related radiation monitoring and sampling systems is performed

under DSRS Chapter 7, and SRP Section 13.3 (Emergency Planning), using RG 1.97 as guidance, radiation monitoring and sampling systems are also used to assess the radiological conditions and consequences associated with AOOs and accidents. For systems that perform non-safety related functions but are used to assess compliance with 10 CFR Part 20 during AOOs and accident conditions, the guidance of RG 1.97 may be used in evaluating instrumentation that perform dual or complementary monitoring functions. For systems designated as safety-related, the review, per RG 1.97, addresses the performance, design, qualification, display, QA, and selection of monitoring variables of radiation monitoring equipment required for accident monitoring and sampling.

Under the provisions of 10 CFR 50.34(f)(2) on TMI-related requirements, applicants must describe provisions for sampling and monitoring process and effluent streams and for conducting analysis of samples, including the proposed analytical programs, during postulated accidents and confirm that such provisions are implemented in accordance with the requirements of 10 CFR 50.34(f)(2)(viii), 10 CFR 50.34(f)(2)(xiv)(E), 10 CFR 50.34(f)(2)(xvii), 10 CFR 50.34(f)(2)(xxvi), and 10 CFR 50.34(f)(2)(xxvii), and the guidelines in RG 1.97.

Moreover, the design of monitoring and sampling systems should meet the standards of NUREG-0718, NUREG-0933, and NUREG-0737 (item II.F.1 and Attachments 1 and 2). Similarly, compliance with GDC 60, 63 and 64 requires that means be provided to (1) control and monitor radioactive materials in effluent releases, (2) monitor waste facilities and radioactive systems for excessive radiation levels, and (3) survey radioactive effluent discharge paths and plant environs for radioactivity released during normal operation, AOOs, and postulated accidents. In both instances, supporting details for the staff's evaluation and staff's findings of the adequacy of the applicant's information are presented in DSRS Chapters 7 and DSRS Sections 9.3.2, 11.5, 12.3-12.4, 12.5, and SRP Section 13.3 and RG 1.52 and 1.140.

To the extent practicable, the same non-safety related instruments should be used for accident monitoring as are used for normal operations of the plant. The staff should compare the application description and assignment of variable types, performance criteria, design criteria (e.g., power supply, calibration, testability etc.), qualification criteria, methods for displaying data and the QA requirements of non-safety related radiation monitoring instrumentation, to the guidance contained within RG 1.97 and IEEE Std 497-2002.

As part of the review, the staff should confirm that provisions have been made for the installation of monitoring instrumentation, sampling, and sample analyses for all identified gaseous effluent release paths in the event of a postulated accident. In addition, the TMI related requirements identify the needs to sample and analyze primary and secondary coolant to determine radionuclide concentrations, radioactivity levels in the reactor building atmosphere, and ambient radiation exposure rates within the operating areas of the reactor building. The review should confirm whether the applicant has proposed administrative controls and procedures to monitor accidents and accidental releases of radioactive gaseous and particulate effluents.

The review should consider the placement and operating characteristics of gaseous, radioiodines, and particulate monitors; the means to locate samples within plant system and on effluent discharge points (e.g. plant stacks and building vents); and identify which monitoring systems have automatic termination of effluent releases into unrestricted areas or diversion of releases in the event that release set-points are exceeded. Whenever two or more radiation monitoring systems are used for accident monitoring on a single discharge point, the applicant

should describe differences in instrumentation response characteristics over their overlapping operational ranges for expected concentrations of noble gases, fission and activation products, and radioiodines.

When evaluating the features of the proposed sampling systems, the staff should confirm that the design includes a means to purge or vent sample lines, and minimize sample losses and distortion in sample chemical and physical compositions. The evaluation should confirm that, once collected, samples are representative of reactor coolant (primary and secondary) and secondary steam. The review should confirm whether the design prevents the blockage of sample lines, and includes restrictions or remotely operated isolation valves to restrict and limit reactor coolant loss in the event that sampling lines rupture. Also, the review should confirm that if inline monitoring is used, the design provides backup provisions for grab sampling, and provides the capability to monitor radiation and radioactivity levels in plant facilities and offsite environs.

The evaluation should review the descriptions of procedures that will be used to operate the radiation monitoring systems. The evaluation should address instrumentation calibration, maintenance, uses of radioactive check sources in confirming operation of instrument channels, sample extraction, and conduct of operational functional checks. Separately, the evaluation should consider the implementation of the quality assurance program and routine inspections of the radiation monitoring systems.

The evaluation should review descriptions of the procedures to ensure that portable instruments are used in the event of an accident and that such instrumentation are capable of measuring the types and levels of expected radiation, with appropriate margins. Given the guidance of IEEE Std 497-2002, the review should confirm that the assigned monitoring variables are consistent with the guidance and included in procedures issued to personnel responding to an emergency.

Radiation Monitoring Systems Subject to Environmental Qualification

For radiation monitoring equipment subject to environmental qualification, under the requirements of 10 CFR 50.49 and GDC 4, the review should confirm that the applicant has identified all appropriate systems and components, and has provided information describing the methods used to qualify equipment located in plant areas with elevated radiation levels. In the context of this DSRS section, the review should consider ambient radiation levels from all sources of radioactivity and the total dose expected during normal operation over the installed life of the equipment, and total dose anticipated during accident conditions.

For plants using Revision 4 of RG 1.97, non-safety related accident monitoring equipment identified as Type A, B, or C variables in accordance with that guide should be environmentally qualified as required by 10 CFR 50.49. Type D variables should be environmentally qualified for the particular accident's postulated environment at the installed location in accordance with the plant's licensing basis. Non-safety related radiation monitoring equipment for which credit was taken in satisfying RG 1.97 should satisfy the seismic qualification guidelines identified in RG 1.97.

RG 1.97 and IEEE Std 497-2002, as referenced above, provide detailed guidance and criteria for non-safety related post accident radiation monitoring instrumentation. The staff

will coordinate the review of the non-safety related 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. The staff should review the methods, models, and assumptions of the applicant's analyses demonstrating that the radiation monitoring instrumentation indications would remain on scale, with appropriate margins, for any design basis event or accident for which the instrumentation might be required for operator information. An appropriate margin should include allowance for analytical and instrumentation uncertainties.

IV. EVALUATION FINDINGS

The reviewer verifies that the applicant has provided sufficient information and that the review and calculations (if applicable) support conclusions of the following type to be included in the staff's safety evaluation report. The reviewer also states the bases for those conclusions.

If appropriate, the staff concludes that the designs of the radiation monitoring and sampling systems (as permanently installed systems or in combination with skid-mounted systems) include the equipment necessary to monitor ambient radiation exposure rates and airborne concentration levels, radioactivity in process and effluent streams, and control releases of radioactive materials associated with the operation of plant systems. The designs are found to be acceptable and meet the applicable requirements of 10 CFR 20.1101(b), 10 CFR 20.1201, 10 CFR 20.1202, 10 CFR 20.1301, 10 CFR 20.1302, 10 CFR 20.1406, and 10 CFR 20.1501; 10 CFR 50.34a, 10 CFR 50.36a, 10 CFR 50.34(f)(2)(viii), 10 CFR 50.34(f)(2)(xiv)(E), 10 CFR 50.34(f)(2)(xvii), 10 CFR 50.34(f)(2)(xxvi), and 10 CFR 50.34(f)(2)(xxvii); 10 CFR 50.49; 10 CFR 50.65(a); 10 CFR 50, Appendix I dose objectives and ALARA provisions; and 10 CFR Part 50, Appendix A GDC 19, 60, 61, 63, and 64.

This conclusion is based on the following:

1. The radiation monitoring system includes instrumentation to monitor and sample radioactivity for contaminated liquid, gaseous, and solid waste process and effluent streams. The staff evaluated the provisions proposed to sample and monitor all appropriate process streams and effluent release points, including nonradioactive systems that could become contaminated through interfaces with radioactive systems. The designs were found to be acceptable and to meet the applicable requirements of GDC 60 and 64.

2. The radiation monitoring system includes instrumentation to monitor ambient radiation exposure rates and means to collect samples in assessing radioactivity concentration levels in plant areas occupied by radiation workers within liquid, gaseous, and waste processing streams. The staff evaluated the provisions for sampling and monitoring appropriate process streams, including nonradioactive systems that could become contaminated through interfaces with radioactive systems. The designs were found to be acceptable and to meet the applicable requirements of 10 CFR 20.1101(b), 20.1201, 20.1202, 20.1301, 20.1302, 20.1406, and 20.1501, using the guidance of RG 8.8 and 8.10.

3. The radiation monitoring system includes provisions for the automatic termination of effluent releases and ensures control over discharges, in accordance with GDC 60.

The provisions proposed for sampling and monitoring liquid, gaseous, and solid waste process streams are in accordance with GDC 63. The provisions for sampling process and effluent streams and conducting analysis of samples, including the proposed analytical programs, were found to be in accordance with the guidelines in RG 1.21, 1.33, 4.8, and 4.15 for routine plant operation and AOOs.

4. The area radiation monitoring system meets the criteria of 10 CFR 50.34(f)(2)(xvii), 10 CFR Part 50, Appendix E, Section VI.2(a), item II.F.1(3) of NUREG-0737, RG 1.97 and Std IEEE Std 497-2002 and is equipped with local and remote audio and visual alarms and a facility for central recording. (*Note to staff: reviewer should list such systems based on information provided by applicant*).

5. The staff's evaluation determined that accident radiation monitoring systems have been provided with the capability to assess radiation hazards in areas that may be occupied by personnel during the course of an accident. The installed instruments have been provided with emergency power supplies, and the portable instruments are placed in locations that are 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, with appropriate margins for the accident source term 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.

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

7. The provisions for sampling and monitoring process and effluent streams and conducting analysis of samples, including the proposed analytical programs, during postulated accidents were found to be in accordance with the requirements of 10 CFR 50.34(f)(2)(viii), 10 CFR 50.34(f)(2)(xiv)(E), 10 CFR 50.34(f)(2)(xvii), 10 CFR 50.34(f)(2)(xxvi), and 10 CFR 50.34(f)(2)(xxvii), and the guidelines of DSRS Section 11.5, and Appendix 11.5-A, as supported by the reviews and evaluations conducted under SRP Section 13.3 and DSRS Chapter 7.

8. The review evaluated P&IDs, process flow diagrams, and descriptions of' proposed sampling points for the liquid, gaseous, and solid waste systems, provisions for local ventilation, and locations of monitoring and sampling points relative to effluent release points, as shown on site plot diagrams. The descriptions of the systems and provisions to collect samples were found to be consistent with NRC guidance and requirements of GDC 60 and 64.

9. The staff reviewed the applicant's QA provisions for the radiation monitoring system, and the quality group and safety classifications assigned to system components, as described in RG 1.143. Compliance with RG 1.143 provides reasonable assurance that components comply with the requirements of GDC 61. The elements of the QA program were found to be consistent with the NRC guidance contained in RG 1.21, 1.33, 4.8, and 4.15; Generic Letter 89-01; and NUREG-1301 and NUREG-0133.

10. The staff reviewed the provisions incorporated in the applicant's design to (1) control releases of radioactive materials in wastes and effluents caused by spills, leaks, and

inadvertent tank overflows; (2) avoid the contamination of nonradioactive systems; (3) prevent uncontrolled and unmonitored releases of radioactive materials in the environment; and (4) avoid interconnections with potable and sanitary water systems. On the basis of this review, the staff concludes that the applicant's proposed measures are consistent with the guidance of RG 1.143 and 4.21 and requirements of GDC 60 and 64.

11. The staff concludes that the RETS/SREC, ODCM, administrative programs, and operational procedures associated with their implementation are consistent with the requirements of Generic Letter 89-01; the guidance of NUREG-1301 and NUREG-0133; and the guidance of RG 1.21, 1.33, 4.8, and 4.15. The staff's review and evaluation of these operational programs and procedures are addressed in DSRS Section 11.5.

12. For liquid and gaseous process radiation monitoring equipment not covered by the ODCM, the review confirmed that the applicant has provided sufficient information describing methods and procedures that will be used in deriving lower limits of detection or detection sensitivities, and set-points (alarms and process termination/diversion) for process radiation monitoring equipment. Similarly, the applicant has provided information describing a plant-specific process and radiological sampling and analysis plan for systems not covered by the ODCM. These provisions describe sampling and analytical frequencies, and radiological analyses for the expected types of liquid and gaseous samples and waste media generated by the LWMS, GWMS, and SWMS and other plant systems (*Note to staff: reviewer should list such systems based on information provided by applicant*). The review confirmed that the proposed sampling and analytical frequencies, and radiological analyses are commensurate with the expected levels of radioactivity in their associated systems.

13. The staff has reviewed the application and determined that the applicant has met the requirements of 10 CFR 20.1406 with respect to describing how the 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.

14. The staff has reviewed the methods and results describing the environmental qualification of equipment exposed to radiation over their operating life and during accident conditions under the requirements of 10 CFR 50.49. The staff finds the methods and results presented by the applicant acceptable in characterizing the radiation or radioactive environments in which equipment is located and the total expected doses during normal operation over the installed life of the equipment, and doses anticipated during accident conditions.

15. In accordance with the requirements of 10 CFR 50.65(a), using the guidance provided in RG 1.160 as endorsed in the application, the applicant has described procedures to monitor the performance or condition of SSCs, including those that are relied upon to mitigate accidents or transients or are used in plant EOPs, against licensee-established goals in a manner sufficient to provide reasonable assurance that SSCs important to safety, including those that are relied upon to mitigate accidents or transients or are used in plant EOPs (e.g., radiation monitors described

in this section and in DSRS Section 11.5 or radiation protection features described in DSRS Section 12.3) are capable of fulfilling their intended functions.

16. 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 instances where an applicant has submitted conceptual design information for portions of the plant for which the application does not seek certification, the findings will summarize the staff's evaluation confirming that the applicant has submitted supplemental design details for the associated SSCs, adequately addressed interface requirements with other SSCs that are included in the design certification, and determined the adequacy of the proposed ITAAC and methods used in verifying that all interface requirements have been met by the COL applicant under the requirements of 10 CFR 52.47(a)(24)–(26), 10 CFR 52.79(d)(2), and 10 CFR 52.80(a).

    In addition, to the extent that the review is not discussed in other safety evaluation report (SER) sections, the findings will summarize the staff's evaluation of the ITAAC, including design acceptance criteria, as applicable.

V. IMPLEMENTATION

The regulations in 10 CFR 52.17(a)(1)(xii), 10 CFR 52.47(a)(9), and 10 CFR 52.79(a)(41) establish requirements for applications for ESPs, DCs, and COLs, respectively. These regulations require the application to include an evaluation of the site (ESP), standard plant design (DC), or facility (COL) against the Standard Review Plan (SRP) revision in effect six months before the docket date of the application. While the SRP provides generic guidance, the staff developed the SRP guidance based on the staff's experience in reviewing applications for construction permits and operating licenses for large light-water nuclear power reactors. The proposed small modular reactor (SMR) designs, however, differ significantly from large light-water nuclear reactor power plant designs.

In view of the differences between the designs of SMRs and the designs of large light-water power reactors, the Commission issued SRM- COMGBJ-10-0004/COMGEA-10-0001, "Use of Risk Insights to Enhance the Safety Focus of Small Modular Reactor Reviews," dated August 31, 2010 (ML102510405) (SRM). In the SRM, the Commission directed the staff to develop risk-informed licensing review plans for each of the SMR design reviews, including plans for the associated pre-application activities. Accordingly, the staff has developed the content of the DSRS as an alternative method for the evaluation of a NuScale-specific application submitted pursuant to 10 CFR Part 52, and the staff has determined that each application may address the DSRS in lieu of addressing the SRP, with specified exceptions. These exceptions include particular review areas in which the DSRS directs reviewers to consult the SRP and others in which the SRP is used for the review. If an applicant chooses to address the DSRS, the application should identify and describe all differences between the design features (DC and COL applications only), analytical techniques, and procedural measures proposed in an application and the guidance of the applicable DSRS section (or SRP section as specified in the DSRS), and discuss how the proposed alternative provides an acceptable method of complying with the regulations that underlie the DSRS acceptance criteria.

The staff has accepted the content of the DSRS as an alternative method for evaluating whether

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

VI. REFERENCES

1. 10 CFR Part 20, "Standards for Protection Against Radiation."

2. 10 CFR 20.1101(b), "Radiation Protection Programs."

3. 10 CFR 20.1201, "Occupational Dose Limits for Adults."

4. 10 CFR 20.1202, "Compliance with Requirements for Summation of External and Internal Doses."

5. 10 CFR 20.1301, "Dose Limits for Individual Members of the Public."

6. 10 CFR 20.1301(e), "Dose Limits for Individual Members of the Public.".

7. 10 CFR 20.1302, "Compliance with Dose Limits for Individual Members of the Public."

8. 10 CFR 20.1406, "Minimization of Contamination."

9. 10 CFR 20.1501, "General.".

10. 10 CFR Part 20, Appendix B, "Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage."

11. 10 CFR Part 50.34, "Contents of Applications; Technical Information.".

12. 10 CFR 50.34a, "Design Objectives for Equipment to Control Releases of Radioactive Material in Effluents—Nuclear Power Plants."

13. 10 CFR Part 50.34(f), "Additional TMI-Related Requirements."

14. 10 CFR 50.36a, "Technical Specifications on Effluents from Nuclear Power Reactors."

15. 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety of Nuclear Power Plants."

16. 10 CFR 50.55a, "Codes and Standards."

17. 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at

Nuclear Power Plants."

18. 10 CFR Part 50, Appendix A, GDC 19, "Control Room."

19. 10 CFR Part 50, Appendix A, GDC 60, "Control of Releases of Radioactive Materials to the Environment."

20. 10 CFR Part 50, Appendix A, GDC 61, "Fuel Storage and Handling and Radioactivity Control."

21. 10 CFR Part 50, Appendix A, GDC 63, "Monitoring Fuel and Waste Storage."

22. 10 CFR Part 50, Appendix A, GDC 64, "Monitoring Radioactivity Releases."

23. Appendix B to 10 CFR Part 50, Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."

24. Section VI. "Emergency Response Data System of Appendix," of 10 CFR Part 50, Appendix E "Emergency Planning and Preparedness for Production and Utilization Facilities."

25. 10 CFR Part 50, Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low as is Reasonably Achievable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents."

26. 10 CFR Part 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants."

27. 10 CFR PART 70, "Domestic Licensing of Special Nuclear Material"

28. 10 CFR Part 100, "Reactor Site Criteria."

29. Generic Letter 89-01, "Implementation of Programmatic Controls for Radiological Effluent Technical Specifications in the Administrative Controls Section of the Technical Specifications and the Relocation of Procedural Details of RETS to the Offsite Dose Calculation Manual or to the Process Control Program (Generic Letter 89-01)," January 31, 1989.

30. SECY-05-0197, "Review of Operational Programs in a Combined License Application and Generic Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria," October 28, 2005.

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

32. RG 1.13, "Spent Fuel Storage Facility Design Basis."

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