U.S. NUCLEAR REGULATORY COMMISSION

DESIGN-SPECIFIC REVIEW STANDARD FOR NuScale SMR DESIGN

11.1 COOLANT SOURCE TERMS

REVIEW RESPONSIBILITIES

Primary - Organization responsible for review of coolant source terms associated with normal operations, and anticipated operational occurrences

Secondary - Organization responsible for review of coolant source terms associated with the evaluation of design basis accidents and equipment qualification

I. AREAS OF REVIEW

At the early site permit (ESP) or standard design certification (DC) stage of review, the staff reviews the information in the applicant's final safety analysis report (FSAR) on the sources of radioactivity that are processed by radioactive waste management systems (RWMS) in treating liquid and gaseous wastes. At the combined license (COL) stage of review, the staff confirms that the information accepted at the standard DC stage is appropriately incorporated in the relevant sections of COL applications, or that proposed departures are adequately justified and documented.

The Design-Specific Review Standard (DSRS) utilizes various source terms for a variety of purposes, including:

1. Normal operation source terms, based on operational reactor experience, as described in American Standard National Institute/American Nuclear Society (ANSI/ANS) N18.1-1999. Addressed here in DSRS Section 11.1 for reactor coolant (primary and secondary) and specific reactor design details, and DSRS Sections 11.2 and 11.3 for system design features used to process and treat liquid and gaseous effluents before being released or recycled.

2. Anticipated operational occurrence (AOO) source terms, based on the more limiting of technical specifications, or design basis source term, used to determine the effects of events like primary to secondary leaks. Addressed here in DSRS Section 11.1 for reactor coolant (primary and secondary) and specific reactor design details.

3. With respect to the immersion of the reactor module in the reactor pool, the review will address production mechanism and concentration of neutron activation products in reactor pool water. In a parallel concern, the review will address whether neutron activation of the secondary coolant is possible given the close proximity of feedwater nozzle plenums to the top of the reactor core. The presence of neutron activation products should reflect target elements present in demineralized water and inadvertent introduction of impurities into pool water and secondary coolant during maintenance, refueling operations, and component failures, e.g., resin breakthrough out of resin traps.

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4. Design basis source terms, e.g., based on 0.25 - 1% fuel defects used to determine shielding and ventilation design requirements in complying with regulations. Addressed in DSRS Section 12.2 for source terms contained in systems and components. This information is also used in DSRS Section 12.2 in developing post-accident shielding (for vital area access, including work areas) source terms in addressing NUREG-0737, Item II.B.2, or the guidance of Regulatory Guide 1.183.

5. Equipment Qualification (EQ) source term, which may or may not be more limiting than the stated accident source term. Addressed in DSRS Sections 3.11 and 12.2 in developing source terms used to assess dose and dose rates to equipment.

6. Accident source terms, which are based on design basis events, are used for determining doses to the public and plant operators during design basis accidents. Addressed in DSRS Chapter 15.

As described below, this DSRS addresses the review of coolant source terms used to evaluate radioactive waste management systems in small integrated pressurized-water reactors (iPWRs); however, this review standard may be applicable to other small modular pressurized-water reactors (PWRs) with similar reactor design features. The review does not address an evaluation of plant and process equipment, neutron-activated components, in-core neutron detectors, or spent-fuel, but relies on plant operating characteristics and RWMS design parameters in calculating radionuclide concentrations in primary and secondary coolant. For the purpose of this DSRS section, radionuclide concentrations in primary and secondary coolant are expected to be representative of operating experience and plant conditions over the life of the plant in estimating radioactivity levels in process and effluent streams. The resulting radionuclide concentrations are not intended to be used as the sole basis for the design of the plant and RWMS.

The design basis coolant source term is used to derive inventories of radioactivity in system components, assess the adequacy of shielding in maintaining doses to workers and public as low as reasonably achievable (ALARA), define ambient radiation exposure levels and zones, and confirm the proper placement of radiation monitoring equipment in plant areas and operating conditions, and the design of ventilation systems provided for maintaining doses to workers ALARA, consistent with Title 10 of the *Code of Federal Regulations* (10 CFR) Part 20, Subparts G and H. The design basis source term is based on a combination of assumptions of failed fuel fractions (e.g., 0.25 to 1 percent) for PWRs, or the reactor coolant system isotopic concentrations, including fission products and significant corrosion and activation products, equivalent to operation for a full fuel cycle at the technical specification limits for halogens (I-131 dose equivalent) and noble gases (Xe-133 dose equivalent), and steam generator technical specification limits on primary to secondary leakage and, as warranted, presence of neutron activation products in secondary coolant.

This information may be used, in part, to support the development of other source terms, such as in framing assumptions for design basis accidents in evaluating radiation doses for equipment qualifications, and radiation protection measures for other materials stored in spent-fuel pools. For these specific applications, the requirements and guidance, and the staff's evaluation process are addressed in DSRS Section 12.2, "Radiation Sources," DSRS Section 15.0.3, "Design Basis Accident Radiological Consequences of Analyses for Advanced Light Water Reactors," and DSRS Section 3.11, "Environmental Qualification of Mechanical and

Electrical Equipment," respectively. The results of the staff's evaluation and acceptability of these source terms and associated system parameters applied in their development will be determined in DSRS Section 12.2 for radiation protection purposes and shielding design, DSRS Section 15.0.3 for design basis accidents, and DSRS Section 3.11 for equipment qualifications.

The coolant source term for normal operation is based on operating experience of plants with similar type of fuels used in large PWRs. The normal operation source terms are used to assess the performance of RWMS and other systems under normal operating conditions and AOOs. The main difference in the two source terms (normal operation versus design basis) is the adjustment made in deriving radionuclide concentrations in primary and secondary coolants. See DSRS Section 12.2 and DSRS Chapter 15 for details on the development of the design basis source terms, and DSRS Section 3.11 on equipment qualification.

The review will consider the following topics

1. The staff's review of the radioactive coolant source terms includes consideration of parameters used to determine the concentration of radionuclides in the reactor coolant; fraction of fission product activity released to the reactor coolant; and concentrations of all non-fission products in the reactor primary and secondary coolant. Small iPWRs generally have systems common to all PWRs, but differ in some aspects, such as the ratio of the mass of the primary coolant system to that of the fuel. Nevertheless, the generation of fission and activation products, fuel enrichment, fuel cladding and defects, presence of radioactivity in primary and secondary coolant, type coolant purification systems used, and RWMS used to process liquid and gaseous wastes are essentially identical to large PWRs, but differ from prior industry assessments due to NuScale™ different design features, such as 2-year refueling cycle, lack of a steam generator blowdown and treatment system, and presence of neutron activation products in the reactor pool. The following sources of radioactivity are considered in the evaluation of effluent releases:

    A. Gaseous wastes (noble gases, radio-iodine, fission and activation products , carbon-14, and tritium), consisting of offgases from the primary coolant, secondary steam; offgases from the main condenser evacuation system; RCS leakage to the reactor containment vessel evacuation system and reactor pool during reactor module transfers, spent fuel pool and fuel handling areas, and turbine building drains; noble gases stripped from the primary coolant during normal operation and at shutdown; presence and concentration of volatile neutron activation products in the reactor pool; and cover and vent gases from tanks and equipment containing radioactive materials. The presence and concentration of radioactive materials in primary coolant is also expected to account for the type of primary coolant chemistry being used, e.g., boron, lithium, and zinc and hydrogen injection, as defined by the applicant.

    B. Liquid wastes (dissolved or entrained noble gases, radio-iodine, fission and activation products , carbon-14, and tritium), consisting of primary coolant processed to remove radioactive materials and, if applicable, to control boron concentration (shim bleed); leakage collected in equipment and floor drains from buildings housing equipment and components that contain radioactive process fluids; presence and concentration of neutron activation products in the reactor pool; condensate demineralizer regenerant solutions; contaminated liquids from anticipated plant operations, such as resin sluices, filter backwashes, decontamination solutions, and sample station drains; and detergent wastes.

C. Liquid wastes (dissolved or entrained noble gases, radio-iodine, fission and activation products, carbon-14, and tritium), consisting of condensate demineralizer discharges and releases based on secondary coolant concentrations expected during normal operations, concentration of neutron activation products in the reactor pool processed by the liquid waste management system, AOOs, and design basis, or at default activity levels or secondary steam leakage rates derived from technical specifications for secondary coolant.

D. With respect to the immersion of the reactor module in the reactor pool, the review will address production mechanism and concentration of neutron activation products in reactor pool water. In a parallel concern, the review will address whether neutron activation of the secondary coolant is possible given the close proximity of feedwater nozzle plenums to the top of the reactor core. In either case, the review will consider reactions associated with thermal and fast neutrons, expected neutron flux density in zones enveloping each reactor vessel, target elements present in reactor pool water (e.g., specs on boron concentrations, demineralizer water quality, and introduction of impurities into pool water and secondary coolant), and equilibrium concentrations of activated radionuclides and their decay chain products. For longer-lived activation products, the review will consider whether the design includes processing equipment, such as liquid and gaseous waste management systems, that will be used to reduce their concentrations in reactor pool water and ambient atmosphere of the reactor building. In characterizing the associated source terms (pool water and secondary coolant), the review will assess whether the applicant has considered the presence of radionuclides other than those listed in NUREG-0017 and PWR-GALE86 or in ANSI/ANS 18.1-1999. The applicant should document the basis for the presence of additional radionuclides, generated by neutron activation, and provide sufficient sufficient information for the staff to conduct an independent evaluation. The presence of neutron activation products should reflect target elements present in demineralized water and inadvertent introduction of impurities into pool water and secondary coolant during maintenance, refueling operations, and component failures, e.g., resin breakthrough out of resin traps.

2. Additional Information for 10 CFR Part 52 Applications: Additional information will be provided by the applicant depending on the type of application being submitted for review. For a COL application, the additional information depends on whether the application references an ESP, a DC, both or neither. Information requirements are prescribed within the "Contents of Application" sections of the applicable subparts to 10 CFR Part 52.

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. The review should ensure that plant design features of the certified design are maintained in the COL application and that, if requested, the Part 52 process for seeking exemptions, changes, and departures is observed in changing Tier 1, Tier 2, and Tier 2* information. Additionally, a COL applicant must address requirements and restrictions (e.g., system interfaces and site parameters) included in the referenced DC and how they are being addressed under plant and site-specific conditions.

4. ESP Application Reviews: For an ESP application, submitted under 10 CFR Part 52, Subpart A, the review is limited to the information forming the basis of the radioactive effluent source terms and doses to offsite receptors, as defined by selected reactor technologies (e.g., based on one certified design, or a plant parameter envelope approach based on two or more certified designs) in bounding radioactive liquid and gaseous effluents for all defined release points. The applicant should provide enough information for the staff to conclude that the application provides a bounding assessment in demonstrating the capability to comply with the regulatory requirements of 10 CFR Part 20 and 10 CFR Part 50, Appendix I design objectives. Accordingly, the reviewer should ensure that physical attributes of the site that could affect the design basis of SSCs (in the context of the reviews conducted under DSRS Sections 11.2 and 11.3) that are important to safety or risk significant are reflected in the site characteristics, design parameters, and conditions stipulated in the ESP, including COL action items.

Review Interfaces

Other DSRS sections interface with this section as follows:

1. The reviewer responsible for review of effectiveness of the radwaste systems will use the primary and secondary coolant concentrations calculated above as inputs in evaluating the performance of the liquid waste management system (LWMS), under DSRS Section 11.2, and the gaseous waste management system (GWMS), under DSRS Section 11.3. The purpose of the evaluation is to determine if these systems can adequately treat primary and secondary coolants such that the associated radioactive liquid and gaseous effluents meet the numerical design objectives of Appendix I to 10 CFR Part 50, and liquid and gaseous effluent concentration limits and Note 4 unity criterion of Appendix B Table 2 to 10 CFR Part 20.

2. The reviewer responsible for review of effectiveness of the radwaste systems will coordinate with the review of radiation protection design features under DSRS Section 12.2, for the selection of primary and secondary coolant concentrations used in the design basis source term.

3. The reviewer responsible for review of effectiveness of the radwaste systems - monitoring instrumentation - will review under DSRS Section 11.5, “Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems,” the monitoring and control provisions for all identified effluent release points. The review will also consider monitoring and sampling methods used for the detection of radioactivity in non-radioactive systems to prevent unmonitored and uncontrolled releases of radioactive

materials to the environment.

II. ACCEPTANCE CRITERIA

Requirements

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

1. 10 CFR Part 20, as it relates to determining the operational source term that is used in calculations associated with potential radioactivity in liquid and gaseous effluents to unrestricted areas. While 10 CFR Part 20 is not applicable to an ESP application, 10 CFR 52.17(a)(1) requires an ESP applicant to provide information to provide enough information for the staff to conclude that the application provides a bounding assessment in demonstrating the capability to comply with the regulatory requirements of 10 CFR Part 20 and 10 CFR Part 50, Appendix I design objectives. The information should describe physical attributes of the site, as relevant to the review conducted under this DSRS section, that could affect the design basis of SSCs that are important to safety or risk significant are reflected in the site characteristics, design parameters, and conditions stipulated in the ESP, including COL action items.

2. 10 CFR Part 50.34(a) and (b), 10 CFR Part 52.47(a)(5), and 10 CFR 52.79(a)(3), and 10 CFR 52.79(a)(1)(i) and (ii), as they relate to the kinds and quantities of radioactive materials expected to be produced and released during normal operations and AOOs to be within the limits of Part 20 and Part 50, Appendix I design objectives.

3. 10 CFR Part 50, Appendix I, as it relates to determining the operational source term that is used in calculations associated with potential radioactivity in liquid and gaseous effluents considered in the context of numerical guides for design objectives and limiting conditions for operation to meet the criterion "as low as is reasonably achievable" for radioactive material contained in light-water reactor effluents. The applicant should provide enough information for the staff to conclude that the application provides a bounding assessment in demonstrating the capability to comply with the regulatory requirements of 10 CFR Part 50, Appendix I design objectives.

4. General Design Criterion 60 (GDC) as it relates to determining the operational source term that is used in calculations associated with potential radioactivity in liquid and gaseous effluents released into unrestricted areas, such that a nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents produced during normal reactor operation and AOOs. While GDC 60 is not applicable to an ESP application, an ESP applicant should provide information characterizing anticipated levels of radioactivity in effluents under 10 CFR 52.17(a)(1).

5. GDC 61 as it relates to the design of facilities and shielding used for the safe storage and handling of radioactive materials and other systems containing radioactivity for the purpose of assessing radiological safety under normal operations and postulated accident conditions.

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.

In general, coolant source terms used as the design basis for expected releases have been found acceptable if these values are determined using models and parameters that are consistent with Regulatory Guide (RG) 1.112, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Nuclear Power Reactors," NUREG-0017 (PWR-GALE86 code), and the guidance provided in ANSI/ANS 18.1-1999 once adjusted to reflect the design features of iPWRs. If neutron activation products are expected in reactor pool water and secondary coolant, the applicant should document the basis for the presence of additional radionuclides, other than those listed in NUREG-0017 and PWR-GALE86 or in ANSI/ANS 18.1-1999, and provide sufficient information for the staff to conduct an independent evaluation.

These models and parameters are based on operating experience with large, existing PWRs. Differences in design features and operating characteristics of iPWRs should be evaluated and used to make specific adjustments to the parameters used in NUREG-0017 and PWR-GALE86 or in ANSI/ANS 18.1-1999. Differences may also exist in the operational configuration and sequence of treatment of waste management systems for various process streams and effluent releases. The use of PWR-GALE86 in place of the earlier PWR-GALE code documented in NUREG-0017 is endorsed by Interim Staff Guidance (ISG), DC/COL-ISG-5 (July 2008). Whenever adjustments are made to parameters used in the PWR-GALE code, applicants should document the basis for adjusted parameters used in the PWR-GALE code to allow the staff to conduct an independent evaluation of the applicant's use of alternate code parameters, given sufficient information.

The relevant RGs and ISG are as follows:

1. RG 1.110, as it relates to the cost-benefit analysis for radioactive waste management systems and equipment.

2. RG 1.112, as it relates to the method of calculating releases of radioactive materials in liquid and gaseous effluents from nuclear power plants.

3. RG 1.140, as it relates to the design, testing, and maintenance of normal ventilation exhaust system air filtration and adsorption units at nuclear power plants.

4. DC/COL-ISG-5 on NUREG-0800, Standard Review Plan (SRP) Section 11.1, "PWR-GALE86 Code for Calculation of Routine Radioactive Releases in Gaseous and Liquid Effluents to Support Design Certification and Combined License Applications."

5. All normal operation and AOO sources of radioactive liquid and gaseous effluents delineated above in Subsection I will be considered.

6. For each source of liquid and gaseous waste considered above in Subsection I.1, and as described in DSRS Sections 11.2 and 11.3, the volumes, concentrations, or release rates of radioactive materials given for normal operation and AOOs should be developed using methods that are consistent with those given in NUREG-0017, PWR-GALE86, or ANSI/ANS 18.1-1999.

   Differences in calculation methods and selection of code parameters chosen because of differences in design and operating features between an iPWR and a large PWR should be documented. The applicant should document the basis of differences, with sufficient supporting information included in the application, to allow the staff to conduct an independent evaluation of the applicant's use of alternate code parameters.

7. Decontamination factors used to reduce gaseous effluent releases to the environment, such as noble gases decay tanks, iodine removal systems, and high-efficiency particulate air (HEPA) filters for building ventilation exhaust systems and reactor building cleanup systems should be consistent with those given in RG 1.140. The reactor building mixing factor and filtration efficiency for internal cleanup systems should be consistent in purpose with NUREG-0017, PWR-GALE86, or ANSI/ANS 18.1-1999, or the basis for different ventilation and exhaust systems and cleanup parameters should be documented. The review should evaluate the types and characteristics of filtration systems and adsorbent media proposed to treat gaseous process and effluent streams, including type of charcoal media (grade, mesh size, and bulk density), number and volume of charcoal decay tanks, dynamic adsorption coefficients for charcoal media and retention times, removal efficiencies for HEPA filtration systems, taking into account the expected physical, chemical, and radiological properties of gaseous process and effluent streams and processing flow rates.

8. Decontamination factors applied to reduce liquid effluent releases to the environment should be consistent with those given in NUREG-0017, PWR-GALE86, or ANSI/ANS 18.1-1999 or the basis for different parameters should be documented and supported with sufficient information included in the application. The review should evaluate the types and characteristics of filtration systems, ion-exchange resins, electro-deionization units, reverse osmosis units, and adsorbent media proposed to treat liquid process and effluent streams, including number and volume of ion-exchange resin column or activated charcoal bed, types and volumes of ion-exchange resins or activated charcoals, removal efficiencies and decontamination factors, taking into account the expected physical, chemical, processing flow rates, and radiological properties of liquid process and effluent streams.

9. RWMS system augmentations used in cost-benefit calculations are consistent with the guidance of RG 1.110. The requirements to conduct a cost-benefit analysis and acceptable cost-benefit ratios in assessing the acceptability of such analyses are given in Section II.D of Appendix I to 10 CFR Part 50. Section II.D of Appendix I requires that the applicant shall include in the radwaste system all items of reasonably demonstrated technology that, when added to the system sequentially and in order of diminishing cost-benefit return, can for a favorable cost-benefit ratio effect reductions in dose to the population reasonably expected to be within 50 miles of the reactor. Liquid and gaseous effluent concentration limits at the boundary of the unrestricted area may not exceed the 10 CFR 20 Appendix B, Table 2, Note 4 unity criterion for radionuclide mixtures.

10. The primary and secondary coolant source terms, used in characterizing liquid and gaseous effluents, result in doses meeting the design objectives in unrestricted areas as set forth in Appendix I to 10 CFR Part 50, Sections II.A to II.C.

11. If neutron activation products are expected in reactor pool water and secondary coolant, the applicant should document the basis for the presence of additional radionuclides, other than those listed in in NUREG-0017 and PWR-GALE86 or in ANSI/ANS 18.1-1999, and provide sufficient information for the staff to conduct an independent evaluation.

12. In evaluating the coolant source terms, the applicant should provide the relevant information in the application as required by 10 CFR 50.34(b)(3), 10 CFR 50.34a, and 10 CFR 52.79(a)(3). The FSAR should include all the basic data listed in Appendix B (PWRs) to RG 1.112 in order to calculate releases of radioactive materials in liquid and gaseous effluents. An acceptable method for satisfying the criteria given in Items 1 through 5 consists of using the PWR-GALE code, once adjusted to reflect the design features of iPWRs. Differences in design features and operating characteristics of iPWRs should be evaluated and used to make specific adjustments to the parameters used in NUREG-0017 and PWR-GALE86 or in ANSI/ANS 18.1-1999. Differences may also exist in the operational configuration and sequence of treatment methods among RWMS equipment for various process streams and in treating effluents prior to being released to the environment. Whenever adjustments are made to parameters used in the PWR-GALE code, applicants should document the basis for adjusted parameters used in the PWR-GALE code to allow the staff to conduct an independent evaluation of the applicant's use of alternate code parameters, given sufficient information. If neutron activation products are expected in reactor pool water and secondary coolant, the applicant should document the basis for the presence of additional radionuclides, other than those listed in NUREG-0017 and PWR-GALE86 or in ANSI/ANS 18.1-1999, and provide sufficient information for the staff to conduct an independent evaluation.

13. The design basis coolant source term is based on a combination of assumptions of failed fuel fractions (e.g., 0.25 to 1 percent) for PWRs, or the reactor coolant system isotopic concentrations, including fission products and significant corrosion and activation products, equivalent to operation for a full fuel cycle at the technical specification limits for halogens (I-131 dose equivalent) and noble gases (Xe-133 dose equivalent). The fuel leakage (failed fuel fraction) used for the design basis source term should be consistent with the guidance of DSRS Section 12.2 and should be based on experience with similar PWR fuels under similar operating conditions, including technical specifications for primary and secondary coolant concentrations.

14. When the applicant's calculational technique or any source term parameter differs from that given in NUREG-0017, PWR-GALE86, or ANSI/ANS 18.1-1999, they should be described in sufficient detail in the application and the basis of the alternate methods and/or parameters used should be provided to allow the staff to conduct an independent evaluation.

Technical Rationale

The technical rationale for application of these acceptance criteria is to define the primary and secondary coolant source terms as precursors in calculating radioactivity levels in liquid and gaseous effluents. In addition, this information is used to assess the adequacy and performance of RWMS in treating process streams and controlling amounts of radioactivity discharged in the environment. The technical rationale for the above considerations is discussed in the following paragraphs:

1. 10 CFR Part 50, Appendix I, provides numerical criteria on offsite individual doses due to liquid and gaseous effluents and air doses (as beta and gamma absorbed dose rates) due to gaseous effluents. It also provides an acceptance criterion for cost-benefit analysis as it relates to population doses due to liquid and gaseous effluents (Section II.D of Appendix I). Conformance with Section II.D of Appendix I demonstrates that the plant design includes all items of reasonably demonstrated technology, when added to RMWS in order of diminishing return, will effect a reduction in releases of radioactive materials and cumulative population doses to ALARA levels.

   The calculations using the PWR-GALE86 computer code and the source term parameters given in NUREG-0017 take into account current technology and the availability of equipment based on that technology to reduce radioactivity levels in liquid and gaseous process streams. The assumptions used in the calculations, based on the performance of such equipment, have an impact on design parameters used in modeling the performance of radwaste management systems reviewed in DSRS Section 11.2, "Liquid Waste Management Systems," and in Section 11.3, "Gaseous Waste Management Systems." However, the PWR-GALE86 code is based on a large PWR, and the proportions and scale of iPWRs may differ substantially. If PWR-GALE86 is modified to effectively model an iPWR, the modifications should be described in sufficient detail that they can be reviewed. If an alternate calculational model is developed, it should also be described in detail, and the sources of all parameters used in the model should be described to allow the staff to conduct an independent evaluation.

   Meeting the coolant source term calculation criteria of DSRS Section 11.1 provides reasonable assurance that the system designs reviewed in DSRS Sections 11.2 and 11.3 will meet the effluent concentration limits in unrestricted areas and the Note 4 unity criterion specified in 10 CFR Part 20 (Appendix B, Table 2), the requirements and ALARA objectives of 10 CFR 50.34a as they relate to the adequacy of design information for radwaste management systems; GDCs 60 and 61 of 10 CFR Part 50, Appendix A; and individual dose limits of 10 CFR Part 50, Appendix I.

2. GDC 60 requires, in part, that the nuclear power unit design include means to control suitably the release of radioactive materials in gaseous and liquid effluents produced during normal reactor operation and AOOs.

GDC 60 specifies that sufficient holdup capacity be provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment. The holdup capacity also provides time to allow the shorter-lived radionuclides to decay before they are further processed or released to the environment. Acceptable holdup times are used in the source term calculations provided in NUREG-0017, PWR-GALE86, or ANSI/ANS 18.1-1999.

Meeting the requirements of GDC 60 provides reasonable assurance that releases of radioactive materials during normal operation and AOOs of radwaste processing systems will not result in offsite radiation doses exceeding the numerical design objectives specified in 10 CFR Part 50, Appendix I, and the effluent concentration limits for unrestricted areas specified in 10 CFR Part 20 (Appendix B, Table 2 and Note 4 unity criterion) for mixtures of radionuclides.

3. GDC 61 specifies that systems which may contain radioactivity be the RWMS be designed with a capability to permit appropriate periodic inspection and testing of components important to safety for the purpose of assuring adequate safety under normal and postulated accident conditions.

   GDC 61 also specifies that systems containing radioactive materials be designed to assure adequate safety under normal operations and postulated accident conditions. NUREG-0017 and PWR-GALE86 or in ANSI/ANS 18.1-1999 describe acceptable methods in determining the inventories of radioactive materials in RWMS components during normal operations and assumed to fail during postulated accidents.

   Meeting this requirement of GDC 61 provides reasonable assurance that the necessary information is available to identify the amounts of radioactive materials contained in RWMS and assess the radiological impacts during postulated accidents, as described in DSRS Sections 2.4.13, 11.2 (Branch Technical Position (BTP) 11-6) and 11.3 (BTP 11-5), and analysis of RG 1.143 in assigning the safety classifications to RWMS for design purposes.

III. REVIEW PROCEDURES

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

1. Selected Programs and Guidance - In accordance with the guidance in NUREG-0800, "Introduction - Part 2: Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: Integral Pressurized Water Reactor Edition" (NUREG-0800 Intro Part 2) as applied to this DSRS Section, the staff will review the information proposed by the applicant to evaluate whether it meets the acceptance criteria described in Subsection II of this DSRS. As noted in NUREG-0800 Intro Part 2, the NRC requirements that must be met by an SSC do not change under the SMR framework.

Using the graded approach described in NUREG-0800 Intro Part 2, the NRC staff may determine that, for certain structures, systems, and components (SSCs), the applicant's basis for compliance with other selected NRC requirements may help demonstrate satisfaction of the applicable acceptance criteria for that SSC in lieu of detailed independent analyses. The design-basis capabilities of specific SSCs would be verified where applicable as part of completion of the applicable ITAAC. The use of the selected programs to augment or replace traditional review procedures is described in Figure 1 of NUREG-0800, Introduction - Part 2. Examples of such programs that may be relevant to the graded approach for these SSCs include:

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

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

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

3. In the review of the mathematical models and parameters given in the application to

calculate primary and secondary coolant concentrations, the reviewer compares parameters and calculations given in the application with the models and parameters given in NUREG-0017, PWR-GALE86, or ANSI/ANS 18.1-1999, modified as necessary to reflect the design and operating parameters of the iPWR. If the application includes models or parameters to estimate reactor coolant concentrations that differ from the guidance, the parameters and calculations used should be substantiated by the applicant. The preferred method of substantiation is by presentation of operating data from similar industry information in justifying the basis for any adjustments taking into account plant specific design features and operating conditions. If neutron activation products are expected in reactor pool water and secondary coolant, the applicant should document the basis for the presence of additional radionuclides, other than those listed in NUREG-0017 and PWR-GALE86 or in ANSI/ANS 18.1-1999. The presence of neutron activation products should reflect target elements present in demineralized water and inadvertent introduction of impurities into pool water and secondary coolant during maintenance, refueling operations, and component failures, e.g., resin breakthrough out of resin traps. The reviewer performs an independent calculation of the primary and secondary coolant concentrations using the guidance provided in ANSI/ANS 18.1-1999, modified as necessary to reflect the parameters of plant-specific conditions.

4. In the calculation, the reviewer will use the applicant's values as given in the application for the following key parameters: design core thermal power level, steam flow rate, mass of primary coolant, mass of liquid in steam generators, and coolant purification methods and flow rates, among others. RG 1.112 (Appendix B for PWRs) and NUREG-0017 provide guidance on plant data needed to develop input parameters for the PWR-GALE86 code. The staff may use alternate parameters for the purposes of assessing whether the applicant's values provide a reasonable level of conservatism in assumptions and results. Note: The source terms referenced in this section are used for both the review of the application and environmental report, and for the staff's preparation of the SER and environmental impact statement.

<u>Review Procedures Specific to 10 CFR Part 52 Application Type</u>

A. <u>Early Site Permit Reviews</u>. Subpart A to 10 CFR Part 52 specifies the requirements applicable to the Commission's review of an ESP application. Information required in an ESP application includes a description of the site characteristics and design parameters of the proposed site.

   For review of an ESP application, staff will evaluate the postulated design parameters associated with the normal operational and AOO source terms. The staff should confirm the approach used by the applicant in developing the annual average liquid and gaseous effluent source terms. For a coolant source term based on a single type of reactor design, the staff will confirm that the applied source term is consistent with that presented in the current revision of the design certification for the selected reactor technology. For a coolant source term based on two or more types of reactor designs, the staff will confirm that the source term, as a plant parameter envelope, is consistent with that presented in the current revision of each DC and conservatively bounding over all expected radionuclides and estimated releases, and demonstrate the capability to comply with the regulatory requirements of 10 CFR Part 20 and 10 CFR Part 50,

Appendix I design objectives.

In the absence of certain circumstances, such as a compliance or adequate protection issue, 10 CFR 52.39 precludes the staff from imposing new site characteristics, design parameters, or terms and conditions on the ESP at the COL stage. Accordingly, the reviewer should ensure that physical attributes of the site that could affect the design basis of SSCs (in the context of the reviews conducted under DSRS Sections 11.2 and 11.3) that are important to safety or risk-significant are reflected in the site characteristics, design parameters, or terms and conditions stipulated in the ESP, including COL action items.

B. Standard Design Certification Reviews. For review of a DC application, the reviewer should follow the above procedures to verify that the design, including requirements and restrictions (e.g., system interfaces and site parameters), set forth in the application meet the acceptance criteria. The reviewer should also consider the appropriateness of identified COL action items. The reviewer may identify additional COL action items; however, to ensure these COL action items are addressed during a COL application, they would need to be added to the DC application.

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

For reviews of a COL application relying on a DC, 10 CFR 52.63 precludes the staff from imposing new requirements on DCs unless it is deemed necessary to bring the certification into compliance with NRC regulations, provide adequate protection of public health and safety, or preserve common defense and security. Accordingly, the reviewer should ensure that plant design features of the certified design are maintained in the COL application and that, if requested, the Part 52 process for seeking exemptions, changes, and departures is observed in changing Tier 1, Tier 2, and Tier 2* information.

IV. EVALUATION FINDINGS

The reviewer verifies that the applicant has provided sufficient information and that the staff's safety review and analysis, as augmented by the application of programmatic requirements in accordance with the staff's technical review approach described in the DSRS Introduction, support conclusions of the following types to be included in the staff's SER. The reviewer also states the bases for those conclusions.

The staff concludes that sufficient information has been provided by the applicant so that the requirements of 10 CFR Part 50, Sections 50.34 and 50.34a have been met. The reviewer responsible for review of effectiveness of radwaste systems will provide a summary statement on the acceptability of source terms used as design parameters for the waste management systems will be made under SER Sections 11.2, "Liquid Waste Management Systems," and 11.3, "Gaseous Waste Management Systems."

The staff concludes that the liquid and gaseous source terms are acceptable and that their use in calculating doses associated with liquid and gaseous effluents will meet the regulatory requirements under 10 CFR Part 20 for effluent concentration and dose limits for members of the public, and 10 CFR Part 50, Appendix I design objectives and ALARA provisions. The review includes the bases of the source terms for both the design basis and normal operations and AOOs. The staff confirmed that the source terms were developed using the guidance provided in RG 1.112, NUREG-0017 and ANSI/ANS-18.1-1999, and that specific adjustments were made in consideration of the specific design and operating features of iPWR reactors. The staff confirmed that the applicant has provided sufficient information in justifying changes in the use of input parameters for iPWR reactors.

The staff concludes that the liquid and gaseous source terms are acceptable and that their use in calculating doses associated with accident conditions will meet the requirements of GDC 61. Meeting this requirement of GDC 61 provides the means to determine the amounts of radioactive materials contained in RWMS and assess the radiological impacts during postulated accidents. The staff determined that the applicant used the method and guidance described in , SRP Sections 2.4.13 and 11.2 (BTP 11-6) and SRP Section 11.3 (BTP 11-5), and analysis of RG 1.143 in assigning the safety classifications of RWMS for design purposes.

For an ESP application, the staff confirms that the applicant has provided enough information for the staff to conclude that the application provides a bounding assessment in demonstrating the capability to comply with the regulatory requirements of 10 CFR Part 20 and 10 CFR Part 50, Appendix I design objectives. The staff's evaluation confirmed that physical attributes of the site that could affect the design basis of SSCs (in the context of this SRP section and SRP Sections 11.2 and 11.3) that are important to safety or risk significant are reflected in the site characteristics, design parameters, and conditions stipulated in the ESP, including COL action items. The staff's findings are presented in SER Sections 11.2 and 11.3 in addressing the requirements of 10 CFR Part 20 for effluent concentration limits and dose limits for members of the public, and 10 CFR Part 50, Appendix I design objectives and ALARA provisions. The staff confirms that the approach used by the applicant in developing the annual average liquid and gaseous effluent source terms is consistent with the identified type of reactor design, as presented in the current revision of the design certification. For a coolant source term based on two or more types of reactor designs, the staff confirmed that the source term, as a plant parameter envelope, is consistent with that presented in the DC or other selected reactor technology and conservatively bounding over all expected radionuclides and estimated releases.

For DC and COL reviews, the findings will also summarize the staff's evaluation of requirements and restrictions (e.g., system interfaces and site parameters) and COL action items relevant to this DSRS section. For reviews of a COL application relying on a DC, the staff's findings confirm that plant design features of the certified design are maintained in the COL application and that, if requested, the Part 52 process for seeking exemptions, changes, and departures has been observed in changing relevant information in Tier 1, Tier 2, and Tier 2* information.

V. IMPLEMENTATION

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

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

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

VI. REFERENCES

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

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

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

4. GDC 61, "Fuel Storage and Handling and Radioactivity Control."

5. GDC 19, "Control Room."

6. GDC 4, "Environmental and Dynamic Effects Design Bases."

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

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

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

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

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

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

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

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

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

16. RG 1.29, “Seismic Design Classification.”

17. RG 1.117, “Tornado Design Classification.”

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

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

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

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

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

23. EPRI, “Cobalt Reduction Guidelines.”

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