U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

12.2 RADIATION SOURCES

REVIEW RESPONSIBILITIES

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

Secondary - None

I. AREAS OF REVIEW

The staff will review the applicant's Preliminary Safety Analysis Report (PSAR) for a construction permit (CP), or Final Safety Analysis Report (FSAR) for an operating license (OL), design certification (DC), or combined license (COL), as it relates to radiation sources in normal operations, anticipated operational occurrences (AOOs), and accident conditions used as the bases for determining the radiation protection design features provided to ensure compliance with the requirements of the commission, including: Maintaining Occupational Radiation Exposure (ORE) as low as is reasonably achievable (ALARA), controlling the concentration of radioactive material in the air for workers; controlling the exposure of members of the public from direct radiation sources.

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The staff also reviews sources of radioactive material used as the bases for determining the design features needed to comply with the requirements: to minimize contamination of the facility and the environment, minimize the generation of waste and facilitate decommissioning, to control radioactive material, and to identify the types and quantities of radioactive material that may affect the qualification of electrical equipment important to safety.

The specific areas of review are as follows:

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

2. Airborne Radioactive Material Sources. The staff will review the description of airborne radioactive material sources in the plant considered in the design of the ventilation systems and used for the design of personnel protective measures and for dose assessment. (SAR Chapter 11 contains the description for airborne sources to be considered for their contribution to the plant effluent releases, through equipment of the radioactive waste management systems or the plant ventilation system.) This description should include a tabulation of the calculated concentrations of radioactive material, by nuclide, expected during normal operation, AOOs, and accident conditions for equipment cubicles, corridors, and operating areas normally occupied by operating personnel. It should also include models and parameters for the calculations (PSAR and updated in the OL FSAR, DC FSAR, or COL FSAR or, for sources not described in a referenced certified design, the COL FSAR).

3. Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC). For DC and COL reviews, the staff reviews the applicant's proposed ITAAC associated with the structures, systems, and components (SSCs) related to this Standard Review Plan (SRP) 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 SRP section. Furthermore, the staff reviews the ITAAC to ensure that all SSCs in this area of review are identified and addressed as appropriate in accordance with SRP Section 14.3.

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

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

Review Interfaces

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

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

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

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

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

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

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

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

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

5.4.13 ISOLATION CONDENSER SYSTEM - as it relates to the types, quantities and associated bases for radioactive fission, activation and corrosion products contained in the piping, heat exchangers and other components of the system.

6.5.1 ESF ATMOSPHERE CLEANUP SYSTEMS - as it relates to the types, quantities and associated bases for radioactive fission, activation and corrosion products within the ventilation filtration media and associated structures.

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

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

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

9.2.6 CONDENSATE STORAGE FACILITIES - as it relates to the types, quantities and associated bases for radioactive fission, activation and corrosion products contained in the system piping, tanks, vessels, filtration media and purification media.

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

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

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

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

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

10.4.6 CONDENSATE CLEANUP SYSTEM - as it relates to the types, quantities and concentrations of radioactive fission, activation and corrosion products within the system piping tanks and vessels, filtration media and purification media, as determined from the bases, models and assumptions described in Chapter 11.

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

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

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

II. ACCEPTANCE CRITERIA

Requirements

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

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

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

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

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

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

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

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

8. 10 CFR 52.47(b)(1), which requires that a DC application contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the U.S. Nuclear Regulatory Commission (NRC) regulations.

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

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

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

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

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

13. 10 CFR 50.34(b)(3), 10 CFR 52.47(a)(5), 10 CFR 52.79(a)(3) and 10 CFR 52.157(e) as they relate to identifying the kinds and quantities of radioactive materials expected to be produced in the operation and the means for controlling and limiting radioactive radiation exposures within the limits set forth in 10 CFR Part 20 of this chapter.

SRP Acceptance Criteria

Specific SRP acceptance criteria acceptable to meet the relevant requirements of the NRC regulations identified above are as follows for the review described in this SRP section. The SRP is not a substitute for the NRC regulations, and compliance with it is not required. However, an applicant is required to identify differences between the design features, analytical techniques, and procedural measures proposed for its facility and the SRP acceptance criteria and evaluate how the proposed alternatives to the SRP acceptance criteria provide acceptable methods of compliance with the NRC regulations.

The following regulatory guides, standards, and NUREGs provide information, recommendations, and guidance and in general describe a basis acceptable to the staff for implementing the requirements of 10 CFR 50.34(b)(3), 10 CFR 52.47(a)(5), 10 CFR 52.79(a)(3) and 10 CFR 52.157(e), 10 CFR 20.1201, 10 CFR 20.1202, 10 CFR 20.1203, 10 CFR 20.1204, 10 CFR 20.1206, 10 CFR 20.1207, 10 CFR 20.1301, 10 CFR 20.1406, 40 CFR Part 190, 10 CFR 20.1801 and 10 CFR 50.49.

1. Regulatory Guide (RG) 1.3[2], as it relates to assumptions used in evaluating gaseous concentrations of radionuclides in containment and plant systems following a loss-of-coolant accident, for boiling-water reactors (BWRs).

2. RG 1.4[2], as it relates to assumptions used in evaluating gaseous concentrations of radionuclides in containment and plant systems following a loss-of-coolant accident, for pressurized-water reactors (PWRs).

3. RG 1.183[3], as it relates to the assumptions used in evaluating the concentrations of radionuclides in containment and plant systems following a loss-of-coolant accident.

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

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

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[2] Regulatory Guides 1.3 and 1.4 provide guidance related to Technical Information Document (TID) 14844, "Calculation of Distance Factors for Power and Test Reactor Sites." This guidance is applicable to a holder of an operating license issued prior to January 10, 1997 or a holder of a renewed license under 10 CFR Part 54 whose initial operating license was issued prior to January 10, 1997. These license holders may voluntarily revise the accident source term.

[3] Regulatory Guide1.183 is applicable to applicants or holders of license issued after January 10, 1997.

6. NUREG-0737, Task Action Plan Item II.B.2, as it relates to the identification of specific post accident sources of radiation in the facility.

7. American National Standards Institute/American Nuclear Society (ANSI/ANS) Standard 18.1, as it relates to the establishment of typical long-term concentrations of principal radionuclides in fluid streams of light-water-cooled nuclear power plants.

8. RG 1.89, as it relates to the determination of radiation dose to certain electrical equipment important to safety as described in 10 CFR 50.49.

9. RG 4.21, as it relates to the identification of SSCs that may contain radioactive material used as the bases for determining the location and types of design features used to comply with the requirements of 10 CFR 20.1406 to minimize contamination of the facility and the environment, minimize the generation of waste and facilitate decommissioning.

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

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

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

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

- An offgas rate of 370 MBq/s (100,000 μCi/s) after a 30-minute delay for BWRs.
- 0.25-percent fuel cladding defects for PWRs.
- Postaccident shielding (for vital area access, including work in the area) source terms from NUREG-0737, Item II.B.2, or RG 1.183.

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

The source term used for determining shielding and ventilation design of PWR components provided for purification of secondary coolant, should consider isotopic concentrations associated with operation at the TSs allowed limits for primary-to-secondary leakage and/or the secondary coolant specific activity concentrations.

The tables of source parameters, which can be placed in SAR Chapter 12 or referenced to SAR Chapter 11, will be acceptable if the accompanying text either in this section or other referenced sections makes it clear how the values are used in a shield design calculation, equipment qualification or in a ventilation system design during normal operation, AOOs as defined in 10 CFR Part 50, Appendix A "General Design Criteria for Nuclear Power Plants," and DBEs, as defined in Section (b)(ii) of 10 CFR 50.49 "Environmental qualification of electric equipment important to safety for nuclear power plants." In addition, the quantities will be acceptable if the specific values given in the tables are consistent with ANSI/ANS Standard 18.1 and RG 1.112 for coolant and corrosion activation products source terms. For PWRs designed for the recycling of tritiated water, tritium concentrations in contained sources and airborne concentrations in the regions specified in item I.2 above should be based on a primary coolant concentration of 1.3 E+5 Bq/gm (3.5 μCi/gm).

Technical Rationale

The technical rationale for application of these acceptance criteria to the areas of review addressed by this SRP section is discussed in the following paragraphs:

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

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

3. Compliance with 10 CFR 50.34(b)(3), 10 CFR 52.47(a)(5), 10 CFR 52.79(a)(3) and 10 CFR 52.157(e) ensures that the kinds and quantities of radioactive materials expected to be produced in the operation are described so that the means for controlling and limiting radiation exposures within the limits set forth in 10 CFR Part 20 can be identified.

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

III. <u>REVIEW PROCEDURES</u>

The reviewer should select material from the procedures described below, as may be appropriate for a particular case.

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

The reviewer will consider whether source strengths, concentrations of airborne radioactivity, and quantitative source descriptions are consistent with the assumptions made and the methods used by the applicant. The reviewer should consider whether the bases for RCS source terms are consistent with relevant industry experience about erosion/corrosion rates, fuel integrity and the primary water chemistry control program (i.e., Electric Power Research Institute (EPRI) "Pressurized Water Reactor Primary Water Chemistry Guidelines") including the High Duty Core Index. EPRI developed the "Utility Requirements Document" for evolutionary and advanced light water reactor designs (URD) based on proven technology of 40 years of commercial U.S. and international light-water reactor (LWR) experience. NUREG-1242 "NRC Review of Electric Power Research Institute's Advanced Light Water Reactor Utility Requirements Document," documented the NRC staff's safety evaluation of the URD. The URD reviewed by the staff in 1992 referenced a number of industry documents, such as NP- 6737, "Cobalt Reduction Guidelines," that provided contemporary operating experience regarding design practices beneficial to reducing ORE by controlling potential sources of radiation. While, the state of technology has advanced since the issuance of the initial URD, the reports referenced within the URD, revised versions of those reports and new reports (e.g., “Overview Report on Zinc Addition in Pressurized Water Reactors – 2004“) related to reducing the amount of radioactive material contained in plant systems, are sources of information that describe the current state of technology that may be used to evaluate design specifications provided to ensure ORE is ALARA, consistent with the guidance in RG 8.8 and the requirements of 10 CFR 20.1003 and 1101(b), and 10 CFR 52.47(a)(22) to ensure that operating insights have been incorporated into the plant design. The reviewer will examine locations of the contained sources relative to shield walls, occupied areas, traffic pathways, inservice inspection points, sampling stations, controls, and other parameters for special situations requiring additional action to ensure that ORE will be ALARA and that the bases for the radiation dose to equipment specified by 10 CFR 50.49 are described. Consistent with the requirements in 10 CFR 20.1406 the reviewer uses the guidance contained in RG 4.21, to identify SSCs that could contain radioactive material. Information about the types and quantities of radioactive material contained within SSCs, is used by the staff to support the review of features provided to comply with 10 CFR 20.1406. Based on the review, the staff may request additional information or ask the applicant to reevaluate the analysis and modify those areas that do not meet the acceptance criteria given in Subsection II of this SRP section.

1. For review of a DC application, the reviewer should follow the above procedures to verify that the design, including requirements and restrictions (e.g., interface requirements and site parameters), set forth in the FSAR meets the acceptance criteria. DCs have referred to the FSAR as the design control document. The reviewer should also consider the appropriateness of identified COL action items. The reviewer may identify additional COL action items; however, to ensure these COL action items are addressed during a COL application, they should be added to the DC FSAR.

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

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

IV. <u>EVALUATION FINDINGS</u>

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.

1 The staff’s review should verify that the SAR and its amendments contain sufficient information, in accordance with the provisions of Section 12.2 of RG 1.70 (Section C.I.12.2 of RG 1.206 for DCs and COLs) and 10 CFR 50.34 (10 CFR 52.47 for DCs or 10 CFR 52.79 for COLs) to arrive at conclusions of the following type, which are

to be included in the staff's Safety Evaluation Report (SER). The report will include a summary of the applicant's submittal, the staff's basis for review and acceptance criteria, and the findings of the review. The following is a brief representation of the evaluation findings:

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

B. The applicant has described a facility that can meet the requirements of 10 CFR 20.1101(b), 10 CFR 20.1201, 10 CFR 20.1202, 10 CFR 20.1203, 10 CFR 20.1204, 10 CFR 20.1206, 10 CFR 20.1207, 10 CFR 20.1301, 10 CFR 20.1406,10 CFR 20.1801, 40 CFR Part 190 and 10 CFR 20.1301(e), 10 CFR 50.49 as they relate to the evaluation of source terms and the related provisions of GDC 4 and 61 in Appendix A to 10 CFR Part 50 and supplemented by the guidance of RG 1.112, RG 1.183, NUREG-0737, and ANSI/ANS Standard 18.1.

C. The applicant has provided a description of contained and airborne radioactivity sources used as inputs for the dose assessment and for shielding and ventilation designs. The applicant also included its assumptions in arriving at quantitative values for these contained and airborne source terms, based on ANSI/ANS-18.1, RG 1.112, GDC 61, 10 CFR 20.1201, 10 CFR 20.1202, 10 CFR 20.1203, 10 CFR 20.1204, 10 CFR 20.1206, and 10 CFR 20.1207 and TSs. For postaccident shielding for vital area access, the applicant used the source terms in NUREG-0737 and RG 1.183.

During power operation, the greatest potential for personnel dose is inside the containment from nitrogen-16, noble gases, and neutrons. Outside the containment, and after shutdown inside the containment, the primary sources of personnel exposure are fission products from fuel clad defects, and activation products, including activated corrosion products. The coolant and corrosion activation product source terms are based on operating experience from reactors of similar design; allowances are included for the buildup of activated corrosion products. Neutron and prompt gamma source terms are based on reactor core physics calculations and operating experience from reactors of similar design. Chapter 11 contains other parameters used, as well as a complete description of the routine operation source term development. The accident source terms are based on the NRC short-term lessons learned recommendation in NUREG-0737 and the alternate accident source term in RG 1.183. The source terms presented are comparable to estimates by other applicants with similar designs.

Almost all of the airborne radioactivity within the plant results from equipment leakage. The applicant has provided a tabulation of the maximum expected routine radioactive airborne concentrations in equipment cubicles, corridors, and operating areas because of equipment leakage. The bases for these leakage calculations are in accordance with RG 1.112. The source terms used to develop these airborne concentration values are comparable to estimates by other applicants with similar designs, and are acceptable.

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 SRP section.

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

V. IMPLEMENTATION

The staff will use this SRP section in performing safety evaluations of DC applications and license applications submitted by applicants pursuant to 10 CFR Part 50 or 10 CFR Part 52. Except when the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the staff will use the method described herein to evaluate conformance with Commission regulations.

The provisions of this SRP section apply to reviews of applications submitted six months or more after the date of issuance of this SRP section, unless superseded by a later revision.

The referenced regulatory guides contain the implementation schedules for conformance to parts of the method discussed herein.

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. 40 CFR Part 190, "Protection of Environment," "Environmental Radiation Protection Standards for Nuclear Power Operations."

5. ANSI/ANS Standard 18.1-1999, "Radioactive Source Term for Normal Operation for Light Water Reactors." [4]

6. EPRI "Pressurized Water Reactor Primary Water Chemistry Guidelines." [5]

[4] Copies of this document may be purchased from the American Nuclear Society, 555 North Kensington Avenue, La Grange Park, Illinois 60526 [phone: 800-323-3044] http://www.new.ans.org/

5 Available from the Electric Power Research Institute at http://www.epri.com/

7. EPRI Report TR-016780, “Utility Requirements Document.”[5]

8. EPRI, “Cobalt Reduction Guidelines.”

9. EPRI, “Overview Report on Zinc Addition in Pressurized Water Reactors – 2004.”

10. GDC 61, “Fuel Storage and Handling and Radioactivity Control.”

11. GDC 19, “Control Room.”

12. GDC 4, “Environmental and Dynamic Effects Design Bases,” NUREG-1242, “NRC Review of Electric Power Research Institute's Advanced Light Water Reactor Utility Requirements Document,” Evolutionary Designs, (Vol. 2, Pt. 1, ADAMS Ascension Number: ML100430013 and, Vol. 2, Pt. 2, ADAMS Ascension Number: ML063620331) and Passive Plant Designs (Vol. 3, Pt. 1, ADAMS Ascension Number: ML070600372 and, Vol. 3, Pt. 2, ADAMS Ascension Number: ML070600373).

13. NUREG-0737, “Clarifications of TMI Action Plan Requirements.”

14. RG 4.21, “Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning.”

15. RG 1.3, “Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors.”

16. RG 1.4, “Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors.”

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

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

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

20. RG 1.89 Revision 1, “Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants.”

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

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

PAPERWORK REDUCTION ACT STATEMENT

The information collections contained in the Standard Review Plan are covered by the requirements of 10 CFR Part 50 and 10 CFR Part 52, and were approved by the Office of Management and Budget, approval number 3150-0011 and 3150-0151.

PUBLIC PROTECTION NOTIFICATION

The NRC may not conduct or sponsor, and a person is not required to respond to, a request for information or an information collection requirement unless the requesting document displays a currently valid OMB control number.

SRP SECTION 12.2
Description of Changes

Section 12.2 "RADIATION SOURCES"

This SRP section affirms the technical accuracy and adequacy of the guidance previously provided in Revision 3, dated March 2007, of this SRP. See ADAMS Accession No. ML070710496.

The technical changes incorporated in Revision 4 include: (1) incorporation of regulatory requirements of 10 CFR 20.1406 and the associated guidance contained in RG 4.21 (ML080500187) concerning the minimization of contamination and radioactive waste generation, (2) incorporation of explicit reference to the regulatory requirement of 40 CFR Part 190 implicitly implied by the existing requirement of 10 CFR 20.1301(e) (3) incorporation of the requirements of 10 CFR 50.49(e)(4) and the guidance of RG 1.89 and RG 1.183 to identify the types and quantities of radioactive material that affect the normal operating environment, (4) updated references to regulatory guidance documents not applicable for applications submitted after 1997, (5) incorporation of the regulatory bases stated in 10 CFR 52.47(a)(22), for including operating experience in the design phase, consistent with the existing requirements of 10 CFR 20.1101(b) and the definition of ALARA, and the guidance contained in RG 8.8, (6) incorporated use of the types of documents discussed in the EPRI URD and the associated Safety Evaluation documented in NUREG-1242, as sources of information available to the staff for assessing the use of the current state of technology to reduce ORE, consistent with the guidance contained in RG 8.8 and the requirements of 10 CFR 20.1101(b). The changes to this SRP Subsection reflect the experience gained by the staff developed during NRC reviews of DC and COL applications completed after Revision 3 of SRP 12.2 was issued.

The technical changes in each SRP section are as follows:

I. AREAS OF REVIEW

1. Added a discussion clarifying the purposes of the review under this SRP section.

2. In order to improve how the staff finds information related to the design features discussed in RG 8.8 Regulatory Position C.2, RG 1.206 Subsection C.I.12.2 and RG 4.21, the discussion under Review Interfaces was expanded to provide guidance to the staff on the sections of the SAR expected to identify the types and quantities of radioactive material in the plant.

II. ACCEPTANCE CRITERIA

1. Added clarification of the requirement to identify sources of direct radiation that could result in exposure to members of the public, consistent with 40 CFR Part 190 as stated in 10 CFR 20.1301(e)

2. Provided clarification to the staff regarding the need to identify the types and quantities of radioactive material that could affect radiation environment of equipment important to safety, required by 10 CFR 50.49(e)(4). This change is consistent with and supports the portion of the application review performed in SRP Subsection 3.11.

3. Added a statement describing requirements in 10 CFR 52.47(a)(22) regarding the use of operating experience.

4. Added a statement describing requirements in 10 CFR 20.1406 regarding minimization of contamination of the facility, the environment, waste generation and facilitation of decommissioning.

5. Removed references to RG 1.3 and RG 1.4, because the guidance contained in these documents is not relevant to review of license applications submitted after 1997.

6. Added a statement regarding the use of the guidance contained in RG 1.89 as it relates to identifying types and sources of radiation in accordance with 10 CFR 50.49(e)(4).

7. Added a statement regarding use of source term information for determining the dose to equipment, consistent with the requirements of 10 CFR 50.49(e)(4).

8. Consistent with the guidance contained in RG 1.206 Subsection C.I.12.2, added clarification to description of the quantity of radiation sources to be identified in the application, to those sources requiring shielding considerations.

9. Added a statement regarding the need to identify sources of neutron and gamma radiation resulting from irradiated material that could result ORE or equipment dose inside containment.

10. Added a statement regarding the need to identify radiation sources that could be expected to result from operation with allowable primary-to-secondary leakage.

11. Added statements regarding the applicability of 10 CFR 50.34(b)(3), 10 CFR 52.47(a)(5), 10 CFR 52.79(a)(3) and 10 CFR 52.157(e) as they relate to identifying the kinds and quantities of radioactive materials expected to be produced in the operation and the means for controlling and limiting radiation exposures within the limits set forth in part 20 of this chapter.

III. <u>REVIEW PROCEDURES</u>

1. Added a discussion regarding the use of the types of documents discussed in the EPRI URD and the associated Safety Evaluation documented in NUREG-1242, as sources of information available to the staff for assessing the use of current state of technology to reduce ORD, consistent with the guidance contained in RG 8.8 and the requirements of 10 CFR 20.1101(b).

2. A statement was added to include the identification of sources that form the bases for radiation dose to equipment as required by 10 CFR 50.49(e)(4).

3. Added a reference to TSs.

IV. EVALUATION FINDINGS

1. Added references to 10 CFR 20.1101(b), 40 CFR Part 190 and 10 CFR 20.1301(e), 10 CFR 50.49, GDC-4, and TSs.

V. IMPLEMENTATION

No Changes

VI. REFERENCES

1. Updated the references to regulatory guidance documents to use those relevant to review of license applications submitted after 1997.

2. Added a number of additional documents that are referred to in Sections II and III. Complete references for these documents have been added to the reference section (i.e., Section VI).