U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

12.3 - 12.4 RADIATION PROTECTION DESIGN FEATURES

REVIEW RESPONSIBILITIES

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

Secondary - None

I. AREAS OF REVIEW

The 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 protection design features, taking into account design dose rates, anticipated operational occurrences, and accident conditions.

The specific areas of review are as follows:

1. Facility Design Features

   A. In the CP PSAR, the DC FSAR or the COL FSAR, the description of equipment and facility design features used for assuring that occupational radiation exposure (ORE) will be as low as is reasonably achievable (ALARA)

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B. The radiation zone designations, including zone boundaries for normal operational (including abnormal operational occurrences), refueling, and accident conditions (based on Regulatory Guides 1.3, 1.4, 1.7, or 1.183) (CP PSAR and updates in the OL FSAR, DC FSAR, or the COL FSAR to the extent that they are not addressed in a referenced certified design.

C. The illustrative examples of facility design features of the equipment, components, and systems, including clearly readable scaled layout and arrangement drawings of the facility showing all source locations and the other design details, requested in Section 12.3.1 of Regulatory Guide (RG) 1.70 (CP PSAR and updates in the OL FSAR) or RG 1.206 (DC FSAR or COL FSAR to the extent that they are not addressed in a referenced certified design); specification of shield wall thicknesses for all shielded spaces on the drawings or in separate tables

D. Information describing the implementation of Regulatory Guide 8.8 guidelines on facility and equipment design and layout, as well as information describing any proposed alternatives (CP PSAR and updates in the OL FSAR, DC FSAR, or the COL FSAR to the extent that they are not addressed in a referenced certified design)

E. Information describing design features that will facilitate eventual decommissioning and minimize, to the extent practicable, contamination of the facility and environment and the generation of radioactive waste in accordance with 10 CFR 20.1406 (CP PSAR and updates in the OL FSAR, DC FSAR, or the COL FSAR to the extent that they are not addressed in a referenced certified design)

2. Shielding

A. The shielding to be provided for each of the radiation sources identified in SAR Chapter 11 and Section 12.2, including the design criteria and the shield material to be used for penetrations and for attenuation of neutrons streaming from the annulus between the reactor pressure vessel and biological shield (CP PSAR and updates in the OL FSAR, DC FSAR, or the COL FSAR to the extent that they are not addressed in a referenced certified design); specification of shield wall thicknesses for all shielded spaces on the drawings or in separate tables (as noted in item I.1.C above)

B. The description of the methods by which the shield parameters were determined, including pertinent codes, assumptions, and techniques used or to be used in the calculations (CP PSAR and updates in the OL FSAR, DC FSAR, or the COL FSAR to the extent that they are not addressed in a referenced certified design)

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

D. Information describing implementation of Regulatory Guides 1.69 and 8.8 (regarding special protective features), and information describing any proposed alternatives (CP PSAR and updates in the OL FSAR, DC FSAR, or the COL FSAR to the extent that they are not addressed in a referenced certified design)

E. Descriptions and location of areas (including the access to and egress from) that personnel may need to access following an accident (10 CFR 50.34(f)(2)(vii) and NUREG-0737, Item II.B.2) (CP PSAR and updates in the OL FSAR, DC FSAR, or the COL FSAR to the extent that they are not addressed in a referenced certified design)

F. Physical layout and composition of plant structures and walls that provide shielding for, and barriers to, high and very high radiation areas such that personnel access to and work within these areas can be controlled in accordance with Regulatory Guide 8.38 (CP PSAR and updates in the OL FSAR, DC FSAR, or the COL FSAR to the extent that they are not addressed in a referenced certified design)

3. Ventilation

A. The description of the personnel protection features incorporated in the ventilation system designs called for in Section 12.3.3 of Regulatory Guide 1.70 (CP PSAR and updates in the OL FSAR) or RG 1.206 (DC FSAR or COL FSAR to the extent that they are not addressed in a referenced certified design)

B. Illustrative examples of personnel radiation protection features of the air cleaning system design (CP PSAR and updates in the OL FSAR, DC FSAR, or the COL FSAR to the extent that they are not addressed in a referenced certified design)

C. Information describing the application of Regulatory Guide 1.52 (particularly Sections C.3.10 and 4.10) and Regulatory Guide 8.8, and information describing any proposed alternatives (CP PSAR and updates in the OL FSAR, DC FSAR, or the COL FSAR to the extent that they are not addressed in a referenced certified design)

4. Area Radiation and Airborne Radioactivity Monitoring Instrumentation

A. The description of the fixed area radiation and continuous airborne radioactivity monitoring instrumentation for normal operation, anticipated operational occurrences, and accident conditions, including the criteria for placement, called for in Section 12.3.4 of Regulatory Guide 1.70 (CP PSAR and updates in the OL FSAR) or RG 1.206 (DC FSAR or COL FSAR to the extent that they are not addressed in a referenced certified design)

B. The criteria and method for obtaining representative in-plant airborne radioactivity concentrations in work areas (CP PSAR and updates in the OL FSAR, DC FSAR, or the COL FSAR to the extent that they are not addressed in a referenced certified design)

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

D. Information describing the implementation of radiation monitoring equipment criteria listed in Regulatory Guides 8.2, 8.8, 8.25, and Revision 3 of Regulatory Guide 1.97 and American National Standards Institute (ANSI) Standard N13.1-1999, and information describing any proposed alternatives (CP PSAR and updates in the OL FSAR, DC FSAR, or the COL FSAR to the extent that they are not addressed in a referenced certified design)

E. Description of the in-containment high-range radiation monitoring capability after an accident, in accordance with 10 CFR 50.34(f)(2)(xvii), item II.F.1.3 of NUREG-0737, and Revision 3 of Regulatory Guide 1.97

F. Description of in plant radiation airborne radioactivity monitoring system in accordance with 10 CFR 50.34(f)(2)(xxvii) and item III.D.3.3 of NUREG-0737

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

H. Description of radiation monitors in areas where special nuclear material is handled or stored in accordance with 10 CFR 50.68

5. Dose Assessment

A. The description of the basis for the dose assessment process, providing detailed information as to expected occupancy of plant radiation areas for each radiation zone, and the estimated annual person-sievert (person-rem) doses associated with major functions, such as operation, Radwaste handling, normal maintenance, special maintenance (e.g., steam generator tube plugging), refueling, and inservice inspection, in accordance with the provisions of Regulatory Guide 8.19 (CP PSAR and updates in the OL FSAR, DC FSAR, or COL FSAR)

B. The description of any additional dose-reducing measures taken as a result of the dose assessment process for specific functions or activities (CP PSAR and updates in the OL FSAR, DC FSAR, or COL FSAR to the extent that they are not described in a referenced certified design)

6. Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC). For design certification (DC) and combined license (COL) reviews, the staff reviews the applicant's proposed ITAAC associated with the structures, systems, and components (SSCs) related to this 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.

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

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

Review Interfaces

None

II. ACCEPTANCE CRITERIA

Requirements

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

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

2. 10 CFR 20.1201, as it relates to occupational dose limits for adults.

3. 10 CFR 20.1201, 10 CFR 20.1202, 10 CFR 20.1203, 10 CFR 20.1204, 10 CFR 20.1701, and 10 CFR 20.1702, as they relate to design features, ventilation, monitoring, and dose assessment for controlling the intake of radioactive materials

4. 10 CFR 20.1301 and 10 CFR 20.1302, as they relate to the facility design features that impact the radiation exposure to a member of the public from noneffluent sources associated with normal operations and anticipated operational occurrences

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

6. 10 CFR 20.1601, 10 CFR 20.1602, 10 CFR 20.1901, 10 CFR 20.1902, 10 CFR 20.1903, and 10 CFR 20.1904, as they relate to the identification of potential sources of radiation exposure and the controls of access to and work within areas of the facility with a high potential for radiation exposure

7. 10 CFR 20.1801, as it relates to securing licensed materials against unauthorized removal from the place of storage

8. General Design Criterion (GDC)19 found in Appendix A to 10 CFR Part 50, as it relates to the provision of adequate radiation protection to permit access to areas necessary for occupancy after an accident, without personnel receiving radiation exposures in excess of 50 millisievert (mSv) (5 rem) to the whole body or the equivalent to any part of the whole body for the duration of the accident in accordance with 10 CFR 50.34(f)(vii)[1]

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

10. GDC 63, as it relates to detecting excessive radiation levels in the facility

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

12. 10 CFR 52.47(b)(1), which requires that a DC FSAR contain the proposed inspections, tests, analyses, and acceptance criteria (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 NRC's regulations;

13. 10 CFR 52.80(a), which requires that a COL FSAR contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.

SRP Acceptance Criteria

Specific SRP acceptance criteria acceptable to meet the relevant requirements of the NRC's regulations identified above are as follows for the review described in this SRP section. The SRP is not a substitute for the NRC's 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, NUREGs, and industry standards provide information, recommendations, and guidance and in general describe a basis acceptable to the staff for implementing the requirements of the regulations identified above:

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

1. Regulatory Guide 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. Regulatory Guide 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. Regulatory Guide 1.7, as it relates to methods for determining gaseous radionuclides in containment following an accident.

4. Regulatory Guide 1.52, as it relates to radiation protection considerations for engineered safety feature (ESF) atmosphere cleanup systems operable under postulated design-basis accident (DBA) conditions, to be designated as "primary systems."

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

6. Regulatory Guide 1.97, Revision 3, as it relates to a method acceptable to the staff for complying with the Commission's regulations to provide instrumentation for radiation monitoring following an accident in a light-water-cooled nuclear power plant.

7. Regulatory Guide 1.183[3], as it relates to the assumptions and methods for evaluating doses to individuals accessing the facility during and following an accident in accordance with NUREG-0737, item II.B.2.

8. Regulatory Guide 8.2, as it relates to general information on radiation monitoring programs for administrative personnel.

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

10. Regulatory Guide 8.10, as it relates to the commitment by management and vigilance by the radiation protection manager and staff to maintain ORE ALARA in accordance with 10 CFR 20.1101(b) and the definition of ALARA in 10 CFR 20.1003 .

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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 license holders issued after January 10, 1997.

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

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

13. Regulatory Guide 8.38, as it relates to the physical controls for personnel access to high and very high radiation areas.

14. NUREG-1430, as it relates to radiation protection considerations in the applicability, format, and implementation of the Babcock and Wilcox Technical Specification package.

15. NUREG-1433, as it relates to radiation protection considerations in the applicability, format, and implementation of the General Electric Technical Specification package.

16. NUREG-1434, as it relates to radiation protection considerations in the applicability, format, and implementation of the General Electric Technical Specification package.

17. NUREG-1432, as it relates to radiation protection considerations in the applicability, format, and implementation of the Combustion Engineering Technical Specification package.

18. NUREG-1431, as it relates to radiation protection considerations in the applicability, format, and implementation of the Westinghouse Technical Specification package.

19. ANSI/ANS-HPSSC-6.8.1-1981, as it relates to criteria for the establishment of locations for fixed continuous area gamma radiation monitors and for design features and ranges of measurement.

20. ANSI N13.1-1999, as it relates to the principles that apply in obtaining valid samples of airborne radioactive materials, and acceptable methods and materials for gas and particle sampling.

21. ANSI/ANS-6.4-1997 (R2004), as it relates to requirements and recommended practices for the construction of concrete radiation shielding structures.

22. Memorandum from Larry W. Camper to David B. Matthews and Elmo E. Collins, dated October 10, 2006, and NUREG/CR-3587, as they relate to the design issues that need to be addressed to meet the requirements of 10 CFR 20.1406.

The specific SRP acceptance criteria are:

1. <u>Facility Design Features</u>

   The acceptability of the facility design features will be based on evidence that the applicant has fulfilled the dose limiting requirements of 10 CFR 20.1201, 10 CFR 20.1202, 10 CFR 20.1203, 10 CFR 20.1204, and 10 CFR 20.1207, as well as the radiation protection aspects of GDC 19 and 61, and 10 CFR 50.34. This includes

evidence that major exposure accumulating functions (maintenance, refueling, radioactive material handling and processing, inservice inspection, calibration, decommissioning, and recovery from accidents) have been considered in plant design and that radiation protection features incorporated into the design will keep potential radiation exposure from these activities ALARA in accordance with 10 CFR 20.1101(b), the definition of ALARA in 10 CFR 20.1003, and Regulatory Guides 8.8 and 8.10. Such features may include (1) the ease of accessibility to work, inspection, and sampling areas, (2) the ability to reduce source intensity, (3) design measures to reduce the production, distribution, and retention of activated corrosion products, (4) the ability to reduce time required in radiation fields, and (5) a provision for portable shielding and remote handling tools. Access control will be judged for acceptability in accordance with the requirements of 10 CFR 20.1601, 10 CFR 20.1602, 10 CFR 20.1901, 10 CFR 20.1902, and 10 CFR 20.1903 or access control alternatives in Standard Technical Specifications (NUREG-1430, NUREG-1431, NUREG-1432, NUREG-1433, and NUREG-1434).

Facility design, to the extent practicable, should minimize the potential for creating a very high radiation area during normal operations, including abnormal operational occurrences (such as dropping a fuel bundle during fuel handling operations). High and very high radiation areas should be remote from normally occupied rooms and corridors such that personnel access to these areas can be controlled in accordance with 10 CFR 20.1601 and 10 CFR 20.1602 and the guidance in Regulatory Guide 8.38. All accessible portions of the spent fuel transfer tube or canal that are capable of having radiation levels greater than 1 gray (Gy) per hour (100 rads per hour) should be shielded during fuel transfer. This shielding should be such that the resultant contact radiation levels are no greater than 1 Gy per hour (100 rads per hour). All accessible portions of the spent fuel transfer tube are clearly marked with a sign stating that potentially lethal radiation fields are possible during fuel transfer. If removable shielding is used for the fuel transfer tubes, it must also be explicitly marked as above. If other than permanent shielding is used, local audible and visible alarming radiation monitors must be installed to alert personnel if temporary fuel transfer tube shielding is removed during fuel transfer operations. Similar precautions should also apply to any other plant radiation source having radiation levels greater than 1 Gy per hour (100 rads per hour).

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

on predictions being made for new plant designs, it is expected that the plant shielding can be designed, the plant can be zoned, and sufficient radiation protection design features can be incorporated, such that individuals in shielded areas would receive a small fraction of the 10 CFR Part 20 limits.

All vital areas, in which radiation may unduly limit personnel occupancy during operations following an accident resulting in a degraded core, should be identified. Personnel access to these areas under accident conditions should be demonstrated in accordance with 10 CFR 50.34(f)(2)(vii), using the methods listed in Section II.B.2 of NUREG-0737. The analysis should consider access to, stay time in, and egress from these vital areas.

2. Shielding

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

The applicant's shielding design is acceptable if the methods are comparable to commonly accepted shielding calculations and if assumptions regarding source terms, cross sections, shield and source geometries, and transport methods are realistic. Labyrinth shielded access ways and penetrations should be used to minimize radiation steaming and scatter around shields. Composition of the shielding material should be selected to minimize, to the extent practicable, the potential for the shield itself to become a radiation source (either from activation of the shield material or production of secondary radiation resulting from interactions with the primary radiation). Effective shield design is essential to meeting the criteria that ORE will be ALARA.

In addition, Regulatory Guide 1.69 and ANSI/ANS-6.4-1997 provide guidance on the fabrication and installation of concrete shields for occupational radiation protection at nuclear power plants. Acceptability of the shield construction will be based on an indication that the guidance of these documents have been implemented in facility construction, or that acceptable alternatives have been proposed. Regulatory Guide 8.8 provides additional acceptance criteria regarding shielding and isolation in radiation protection design.

3. Ventilation

The ventilation system will be acceptable for radiation protection purposes if the criteria and bases for ventilation rates within the areas covered in SAR Section 12.2.2 will ensure that air will flow from areas of low potential airborne radioactivity to areas of higher airborne radioactivity and then to filters or vents, that the concentrations of radioactive material in areas normally occupied can be maintained in accordance with the requirements 10 CFR 20.1701, and that the dose limits of 10 CFR 20.1201 are met consistent with the requirements of 10 CFR 20.1202, 10 CFR 20.1203, and 10 CFR 20.1204. The system has adequate capability to reduce concentrations of airborne radioactivity to 1.0 derived air concentration (DAC), as specified in Appendix B to 10 CFR Part 20, in areas not normally occupied where maintenance or inservice inspection must be performed. The system is designed so that filters containing radioactivity can be easily maintained and will not create an additional radiation hazard to personnel maintaining them, or those in adjacent occupied areas. Acceptability of the ventilation system, relative to radioactive gases and particulates, will also be based on evidence that the applicant has applied the guidance of Regulatory Guide 8.8 or proposed acceptable alternatives.

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

4. Area Radiation and Airborne Radioactivity Monitoring Systems

A. The area radiation monitoring systems will be acceptable if they meet the provisions of 10 CFR 20.1501, 10 CFR 50.34(f)(2)(xvii); the guidance in NUREG-0737, Regulatory Guide 8.25, and Regulatory Guide 1.97, Revision 3; and the following criteria:

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

i. Personnel have the capability to assess the radiation hazard in areas that may be accessed during the course of an accident, in accordance with the criteria of 10 CFR 50.34(f)(2)(xvii); NUREG-0737, item II.F.1; and Regulatory Guide 1.97, Revision 3.

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

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

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

v. Two high-range radiation monitors are provided in containment in accordance with the requirements of 10 CFR 50.34(f)(2)(xvii) and item II.F.1 of NUREG-0737.

D. Appendix A to Regulatory Guide 1.21 provides useful guidance about effluent monitoring that applies to the acceptability of in-plant airborne radioactivity monitoring. Regulatory Guide 8.2 includes guidance on surveys to evaluate radiation hazards. The detailed guidance in ANSI N13.1-1999 covers the sampling of airborne radioactive materials in ventilation ducts and stacks of nuclear facilities and may be used for acceptance criteria on the actual sampling process and certain techniques involved. Regulatory Guide 8.8 provides further guidance on monitoring systems.

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

5. Dose Assessment

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

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

2. The referenced sections of 10 CFR Part 50 and 10 CFR Part 52, and the associated sections of Regulatory Guide 1.70 and RG 1.206, specify the scope of the material in an application and the associated technical review by the staff.

3. The references to the specific items in 10 CFR 50.34(f), their associated action item in NUREG-0737, and Regulatory Guide 1.97, Revision 3, specify that adequate in-plant radiation monitoring is provided for accidents and abnormal operational occurrences. Radiation protection design features are provided to allow personnel access to the plant under accident conditions sufficient to perform actions necessary to mitigate the consequences of the accident.

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

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

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

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

III. REVIEW PROCEDURES

The reviewer will 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.

1. The staff will review the information on radiation protection design features furnished in the SAR, including referenced parts of Chapters 9 and 11, for completeness in accordance with Regulatory Guide 1.70 (or RG 1.206 for DC or COL applicants under 10 CFR Part 52). The reviewer will evaluate the SAR text and the scaled layout drawings of the facility, concentrating on the sources, shielding, and layouts for the auxiliary building, including the radwaste systems, decontamination facilities, office and access control areas, laundry, lockers and shower rooms (including the personnel

decontamination area), and laboratory facilities; the fuel handling facilities, including the spent fuel pool fuel transfer and related equipment; and the BWR turbine building, including location of steam lines, reheaters, and moisture separators. For the CP PSAR, this review is particularly concerned with preliminary design features that may not appear to be consistent with ensuring that ORE will be ALARA. This review will evaluate the radiation protection design features using the guidelines of Regulatory Guide 8.8. The reviewer will consider plant layout and intended access and egress traffic patterns both to determine conformance with 10 CFR 20.1601, 10 CFR 20.1602, 10 CFR 20.1901, 10 CFR 20.1902, 10 CFR 20.1903, 10 CFR 20.1904, 10 CFR 20.1905, or Standard Technical Specifications and to determine whether they will control access properly in limited and restricted access areas (high radiation and very high radiation areas). The staff will review SAR Chapters 9 and 11 as necessary to evaluate dose rates in and around the spent fuel pool areas, the location of airborne radioactivity monitoring instruments within ventilation systems, and radwaste systems as they relate to radiation protection design. The reviewer will evaluate all relevant aspects of the initial design plans, particularly to identify new arrangements, improved designs, unusual shield thicknesses, a new or modified shield thickness calculational procedure, unusual assumptions in the calculation, and placement of radiation monitors. The staff responsible for the review of SRP Chapter 11 will evaluate the adequacy of the process and effluent radiation monitoring (e.g., sensitivity, range, system placement) design.

2. Regulatory Guide 1.97, Revision 3, as referenced above, provides detailed guidance and criteria for postaccident radiation monitoring instrumentation. Revision 4 to the regulatory guide provides a method for applying this guidance to the specific proposed design. The staff will coordinate the review of the radiation monitoring systems with the instrumentation and control and emergency preparedness review staff to ensure that adequate radiation detection instrumentation is provided for plant monitoring under accident conditions.

3. The health physics staff will evaluate the adequacy of the applicant's shielding design on the basis of acceptable radiation shielding practices and calculation methods. Based on its review of the plant layout drawings and radiation zoning, the health physics staff may verify, by independent calculations, the adequacy of the shielding design for selected areas of the plant. The review should emphasize areas in the plant that have a potential to become a significant high radiation area (greater than 1 Gy (100 rads) per hour) or a very high radiation area during operations and anticipated operational occurrences. These areas include, but are not limited to, those exposed to gamma shine from steam components in BWR designs (both onsite occupational and offsite public exposure concerns); areas providing access to the spent fuel transfer tube during fuel transfer; the below-vessel reactor cavity, in certain PWR designs, with in-core thimble tubes withdrawn; and the upper drywell, in BWR designs, during fuel movements. Appendix B to Regulatory Guide 8.38 includes guidance on some of these areas.

4. For the OL FSAR, the reviewer will consider any changes in the design that might necessitate changes in operating procedures to accommodate a changed radiation zone or a different location of equipment.

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

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

7. 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 final safety analysis report (FSAR) meets the acceptance criteria. DCs have referred to the FSAR as the design control document (DCD). The reviewer should also consider the appropriateness of identified COL action items. The reviewer may identify additional COL action items; however, to ensure these COL action items are addressed during a COL application, they should be added to the DC FSAR.

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

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

IV. EVALUATION FINDINGS

The reviewer verifies that the applicant has provided sufficient information and that the 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.

In accordance with the provisions of Sections 12.3 and 12.4 of Regulatory Guide 1.70 (or equivalent sections in RG 1.206 for DC or COL FSARs under 10 CFR Part 52) and the radiation protection aspects of 10 CFR 50.34 (or 10 CFR 52.47 or 10 CFR 52.79), as well as radiation protection aspects of GDC 19 and 61, the SAR and amendments provide the basis for conclusions of the following type, which will be included in the staff's safety evaluation report (SER). The report will include a summary of the applicant's coverage, the staff's basis for review and acceptance criteria, and the findings of the review. The following is a brief representation of typical evaluation findings:

> The staff concludes that the radiation protection design features are acceptable and meet the relevant requirements of 10 CFR Part 20, 10 CFR Part 50, GDC 19 and 61, and 10 CFR Part 70. This conclusion is based on the following.

The radiation protection design features at [plant name] are intended to help maintain occupational radiation exposures within regulatory limits and as low as is reasonably achievable, consistent with 10 CFR 20.1101(b) and the definition of ALARA in 10 CFR 20.1003, as well as Regulatory Guides 8.8, and 8.10, the dose-limiting provisions of 10 CFR 20.1201, 10 CFR 20.1202, 10 CFR 20.1203, and 10 CFR 20.1204, and the noneffluent limits in 10 CFR 20.1301 and 10 CFR 20.1302. In addition, the design features are consistent with the radiation exposure and radiation source control requirements in 10 CFR 20.1406, 10 CFR 20.1601, 10 CFR 20.1602, 10 CFR 20.1801, 10 CFR 20.1901, 10 CFR 20.1902, and 10 CFR 20.1905. Many of these design features have been incorporated as a result of the applicant's radiation protection design review and from radiation exposure experience gained during the operation of other nuclear power plants. [Include examples of design features incorporated to reduce radiation to workers during maintenance operations, reduce radiation sources where operations must be performed, allow quick entry and easy access, provide remote operation capability or reduce the time required for work in radiation fields, and examples of other features that reduce radiation exposure of personnel.] These design features are consistent with those contained in Regulatory Guides 8.8 and 8.38 and are acceptable.

Plant design and layout facilitates the control of access to and work within plant areas in accordance with the requirements of 10 CFR 20.1601,10 CFR 20.1602, 10 CFR 20.1901, 10 CFR 20.1902, and 10 CFR 20.1903 and access control alternatives in the Standard Technical Specifications (NUREG-1430, NUREG-1431, NUREG-1432, NUREG-1433, and NUREG-1434) and are acceptable.

Areas within the restricted area are divided into [number of zones] radiation zones. The dose rate criterion for each of these zones is derived from expected occupancy and access restrictions. These criteria are then used as the basis for the radiation shielding design. This allows for arrangements of radioactive equipment that are in accordance with the requirements of 10 CFR Part 20 and the guidelines of Regulatory Guide 8.8. The plant design and layout facilitates the control of access to and work within plant areas in accordance with the requirements of 10 CFR 20.1601, 10 CFR 20.1602, 10 CFR 20.1901, 10 CFR 20.1902, and 10 CFR 20.1903 and access control alternatives in the standard technical specifications (NUREG-1430, NUREG-1431, NUREG-1432, NUREG-1433, and NUREG-1434) and are acceptable.

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

The radiation shielding will be designed to provide protection against radiation for operating personnel, both inside and outside the plant, and for the general public. The following are several of the shielding design features incorporated

into [plant name]. [List several examples of shielding design features used at plant.] Some of the criteria used by [utility] in locating penetrations in shield walls at [plant name] are [list several shield penetration location criteria used]. These shielding techniques are designed to maintain personnel radiation exposures ALARA, in accordance with the provisions of Regulatory Guides 8.8 and 8.10, and are acceptable.

The general shield design methodology and source term inventories used for [plant name] are similar to those from operating reactors. The basic radiation transport analysis used for the applicants' shield design is based on [list appropriate shielding computer codes used]. The applicant also used shielding information from operating nuclear plants as input data for the shield design calculations. All concrete shielding in the plant will be constructed in general compliance with Regulatory Guide 1.69. The staff finds the shielding design and methodology presented in the [CP PSAR, OL FSAR, DC FSAR, or COL FSAR] acceptable based on the SRP criteria.

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

The applicant's area radiation monitoring system is designed to (1) monitor the radiation levels in areas where radiation levels could become significant and where personnel may be present, (2) alarm when the radiation levels exceed preset levels to warn of increased radiation levels, and (3) provide a continuous record of radiation levels at key locations throughout the plant. To meet these objectives, the applicant plans to use [number] area monitors located in areas where personnel may be present and where radiation levels could become significant. The area radiation monitoring system meets the criteria of 10 CFR 50.34(f)(2)(xvii), item II.F.1(3) of NUREG-0737, and Regulatory Guide 1.97, Revision 3, and is equipped with local and remote audio and visual alarms and a facility for central recording. [List examples of other area monitoring system features.] The design objectives of the applicants' airborne radioactivity monitoring system are (1) to assist in maintaining occupational exposure to airborne contaminants ALARA, (2) to check on the integrity of systems containing radioactivity, and (3) to warn of unexpected release of airborne radioactivity to prevent inadvertent exposure of personnel. The applicant will install airborne radioactivity monitors in work areas where there is a potential for airborne radioactivity. These airborne radioactivity monitors have the capability to detect derived air concentrations in air (DAC) of the most restrictive particulate and iodine radionuclides in the area or cubicle of

lowest ventilation flow rate within 10 hours(s) (usually denoted as 10 DAC-hrs). The applicant will provide portable continuous air monitors when needed to monitor air in areas not provided with fixed airborne radioactivity monitors. All airborne and area radioactivity monitors will be calibrated periodically. [List examples of other airborne radioactivity monitoring features.] The objectives and location criteria of [plant name] area and airborne radiation monitoring systems are in conformance with those portions of 10 CFR 20.1501; 10 CFR 50.34, 10 CFR 52.47, or 10 CFR 52.79; and 10 CFR 50.68, as well as Regulatory Guide 1.97, Revision 3, and Regulatory Guide 8.8, related to radiation and airborne radioactivity monitoring.

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

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

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

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

For DC and COL reviews, the findings will also summarize the staff's evaluation of requirements and restrictions (e.g., interface requirements and site parameters) and COL action items relevant to this 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 and NUREGs 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. Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors."

5. Regulatory Guide 1.4, "Assumptions Used for Evaluation of the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors."

6. Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of Coolant Accident."

7. Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Absorption Units of Light-Water-Cooled Nuclear Power Plants."

8. Regulatory Guide 1.69, "Concrete Radiation Shields for Nuclear Power Plants."

9. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants."

10. Regulatory Guide 1.97, Revision 3, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident."

11. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors."

12. Regulatory Guide 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable."

13. Regulatory Guide 8.10, ”Operational Philosophy for Maintaining Occupational Radiation Exposures as Low as Is Reasonably Achievable.”

14. Regulatory Guide 8.19, “Occupational Dose Assessment in Light-Water Reactor Power Plants Design Stage Man-Rem Estimates.”

15. Regulatory Guide 8.25, “Air Sampling in the Workplace.”

16. Regulatory Guide 8.38, “Control of Access to High and Very High Radiation Areas of Nuclear Plants.”

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

18. NUREG-1430, “Standard Technical Specifications for Babcock and Wilcox Plants.”

19. NUREG-1431, “Standard Technical Specifications for Westinghouse Plants.”

20. NUREG-1432, “Standard Technical Specifications for Combustion Engineering Plants.”

21. NUREG-1433, “Standard Technical Specifications for General Electric Plants, BWR/4.”

22. NUREG-1434, “Standard Technical Specifications for General Electric Plants, BWR/6.”

23. ANSI/ANS-HPSSC-6.8.1-1981, “Location and Design Criteria for Area Radiation Monitoring Systems for Light Water Nuclear Reactors.”

24. ANSI/HPS N13.1-1999, “Sampling and Monitoring Releases of Airborne Radioactive Substances from the Stacks and Ducts of Nuclear Facilities.”

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

26. Memorandum from Larry W. Camper to David B. Matthews and Elmo E. Collins, “List of Decommissioning Lessons Learned in Support of the Development of a Standard Review Plan for New Reactor Licensing” (ADAMS Accession No. ML0619201830), October 10, 2006, and NUREG/CR-3587, “Identification and Evaluation of Facility Techniques for Decommissioning of Light Water Reactors.”

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

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