Safety Evaluation Report With Open Items

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12 RADIATION PROTECTION

This chapter describes the U.S. Nuclear Regulatory Commission (NRC) staff review of the radiation protection measures employed by the U.S. EPR, including estimated radiation exposures to plant personnel. This chapter also provides information on facility and equipment design and programs used to meet the radiation protection standards of Title 10 of the *Code of Federal Regulations* (10 CFR), Part 20, "Standards for Protection Against Radiation," Part 50, "Domestic Licensing of Production and Utilization Facilities," and Part 70, "Domestic licensing of Special Nuclear Material."

The NRC evaluated the information in Chapter 12 of the U.S. EPR Final Safety Analysis Report (FSAR) against the guidance in Chapter 12 of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants – LWR Edition" (SRP). Compliance with these criteria provides assurance that doses to workers will be maintained within the occupational dose limits of 10 CFR Part 20. These occupational dose limits, applicable to workers at NRC-licensed facilities, restrict the sum of the external whole-body dose (deep-dose equivalent) and the committed effective equivalent doses resulting from radioactive material deposited inside the body (deposited through injection, absorption, ingestion, or inhalation) to 50 millisievert (mSv) (5 rem) per year with a provision (i.e., by planned special exposure) to extend this dose to 100 mSv (10 rem) per year with a lifetime dose limit of 250 mSv (25 rem) resulting from planned special exposures.

The SRP acceptance criteria also provide the guidance for assuring that radiation doses resulting from exposure to radioactive sources both outside and inside the body can be maintained well within the limits of 10 CFR Part 20 and as low as is reasonably achievable (ALARA). The balancing of internal and external exposure necessary to ensure that the sum of the doses is ALARA is an operational concern. An applicant seeking a combined license (COL) must address these operational concerns, as well as programmatic radiation protection concerns.

12.1 Ensuring that Occupational Radiation Exposures are ALARA

12.1.1 Introduction

ALARA (acronym for "as low as is reasonably achievable") means making every reasonable effort to maintain exposures to radiation as far as practicable below the dose limits of 10 CFR Part 20. This includes taking into account the state of technology and the economics of improvements in relation to benefits to the public health and safety. It also includes using procedures and engineering controls based upon sound radiation protection principles.

In addition to providing radiation exposure limits for workers and members of the public, 10 CFR 20.1101 requires that, to the extent practical, procedures and engineering controls based on sound radiation protection principles be employed to achieve occupational doses and doses to the public that are ALARA. In addition, 10 CFR 20.1704(a) requires that the intake of airborne radioactive materials be consistent with maintaining total effective dose equivalent ALARA. Regulatory Guide (RG) 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable," provides specific guidance and criteria on the design, construction, and operation of a nuclear power plant to meet this regulatory requirement. Programmatic and policy considerations associated with plant operations that are needed to assure that radiation doses will be ALARA (as discussed in RGs 8.8; 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as Is Reasonably Achievable"; and RG 1.8, "Qualification and Training of Personnel for Nuclear Power Plants") are outside the scope of this design certification. The applicant has identified a COL information item (see Section 12.1.5 below) to ensure that COL applicants referencing the design will address these issues.

12.1.2 Summary of Application

FSAR Tier 1: There are no FSAR Tier 1 entries for FSAR Tier 2, Section 12.1, "Occupational ALARA" areas of review.

FSAR Tier 2: The applicant has provided an FSAR Tier 2 system description in Section 12.1, summarized here in part, as follows:

The majority of occupational worker radiation exposure results from maintenance on systems that contain radioactive material, radioactive waste handling, inservice inspection, refueling, abnormal operations, and decommissioning activities. These activities are addressed and included in the design of the U.S. EPR through the plant physical layout, selection of materials, shielding, and chemistry control.

During the design process, the applicant's staff receives guidance on incorporating ALARA into the design. This includes information on lessons learned from the nuclear power industry and from Federal guidance. Proposed design changes undergo review by a design review board. These design reviews include an ALARA review if radiation exposure is affected.

One factor in the cumulative annual facility dose is the facility layout, particularly the proximity of components to each other. An example stated in the facility application relates to the Safeguard Buildings. Each building contains one of four separate equipment divisions. Physical separation and shielding allows an entire train of equipment to be taken out of service, flushed, and maintained while workers are shielded from radioactive material in the other three trains. In addition, higher radioactive sources are confined to the lower floors of each Safeguards Building in which they are located and are placed on one side of the building.

Systems that are designed to contain radioactive materials are modeled in a 3-D computer model from which structural drawings, piping and instrumentation drawings, and isometric drawings are developed. Field-run piping within the Nuclear Island part of the facility is minimized.

The design avoids the use of cobalt-containing alloys for the material selected for components, which minimizes the production, distribution, and retention of activated corrosion products through the primary system. Reactor coolant chemistry control is used to provide a constant pH in a target range that is optimized to minimize the production of corrosion products.

ITAAC: There are no inspections, tests, analyses, and acceptance criteria (ITAAC) associated with FSAR Tier 2, Section 12.1.

Technical Specifications: There are no technical specifications for FSAR Tier 2, Section 12.1.

12.1.3 Regulatory Basis

The relevant requirements of the Commission regulations for ensuring that occupational exposures are ALARA, and the associated acceptance criteria, are given in Section 12.1 of NUREG-0800 and are summarized below. Review interfaces with other SRP sections can also be found in Section 12.1, “Assuring that Occupational Radiation Exposures are as Low as is Reasonably Achievable,” of NUREG-0800.

1. 10 CFR 20.1101 and the definition of ALARA in 10 CFR 20.1003, as they relate to those measures that ensure that radiation exposures resulting from licensed activities are below specified limits and ALARA

Acceptance criteria adequate to meet the above requirements include:

1. RG 1.8 “Qualification and Training of Personnel for Nuclear Power Plants,” as it relates to the qualifications and training of radiation protection personnel

2. RG 8.8 “Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable,” as it relates to providing radiation protection information to ensure that occupational radiation exposure is kept as low as reasonably achievable

3. RG 8.10 “Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as Is Reasonably Achievable” as it relates to commitment by the applicant’s management and vigilance by the radiation protection manager and the radiation protection staff to maintain occupational radiation exposure ALARA

4. RG 8.27, “Radiation Protection Training for Personnel at Light-Water-Cooled Nuclear Power Plants,” as it relates to instructing personnel involved in licensed activities regarding their role and responsibilities for making every reasonable effort to maintain radiation exposures ALARA

5. NUREG-1736, “Consolidated Guidance: 10 CFR Part 20 - Standards for Protection Against Radiation,” as it relates to the requirements for a radiation protection program to maintain doses ALARA

12.1.4 Technical Evaluation

The staff reviewed the information in FSAR Tier 2, Section 12.1, to assess adherence to the guidelines in RG 1.206, “Combined License Applications for Nuclear Power Plants,” Revision 1, as well as the criteria in Section 12.1 of the SRP regarding the radiation protection aspects of the reactor design. Specifically, the staff reviewed FSAR Tier 2, Section 12.1 to ensure that the applicant had either committed to adhere to the guidance of the RGs and staff positions, referenced in Section 12.1 of the SRP, or had provided acceptable alternatives. As described below, the staff finds that FSAR Tier 2, Section 12.1 conforms to the applicable guidance contained in these RGs and applicable staff positions. Therefore, the staff concludes that the relevant requirements of 10 CFR Part 20 have been met.

Policy Considerations

In FSAR Tier 2, Section 12.1.1, “Policy Considerations,” the applicant described the design, construction, and operational policies that have been implemented to ensure that ALARA considerations are factored into each stage of the design process. The applicant has committed to ensure that the plant will be designed and constructed in a manner consistent with the guidelines of RG 8.8. In particular, FSAR Tier 2, Section 12.1.2, “Design Considerations,” states that the applicant has met this commitment by training designers and engineers on incorporating ALARA into the design evolution process. This training included communicating review responsibilities, as well as communicating lessons learned from the nuclear power industry and Federal guidance. Design changes which affected radiation exposure underwent an ALARA board review, with board members selected from U.S., German, or French design teams, as well as operators from existing U.S. nuclear power plants. In addition, structural, piping, instrumentation, and isometric drawings of radioactive systems were developed using a 3-D computer model in order to avoid radiation streaming, verify smooth piping runs and laminar flow, ensure proper piping slope, avoid crud traps, verify the location of low point drains, verify the location of high point vents, and verify the segregation of radioactive and nonradioactive piping. This ALARA design and construction policy conforms to the applicable guidelines of RG 8.8 and is, therefore, acceptable.

The requirements of 10 CFR Part 20 specify that all licensees must develop, document, and implement a radiation protection program. Specifically, this program shall encompass the ALARA concept and include provisions for maintaining radiation doses and intakes of radioactive materials ALARA. The operational ALARA policy forms the basis for the operating station’s ALARA manual. The detailed policy considerations regarding overall plant operations and implementation of such a radiation protection program are outside the scope of the design certification review.

In order to maintain doses to plant personnel ALARA, the applicant stated, in FSAR Tier 2, Section 12.1.3, “Operational Considerations,” that the COL applicant will submit a description of an ALARA program which will comply with the requirements of 10 CFR Part 20 and be consistent with the applicable portions of NUREG-1736 and the guidance in the following regulatory guides. This is specified in FSAR Tier 2, COL Information Item 12.1-1.

- RG 1.8, Revision 3
- RG 8.2, “Guide for Administrative Practices in Radiation Monitoring”
- RG 8.7, Revision 2, “Instructions for Recording and Reporting Occupational Radiation Exposure Data”
- RG 8.8, Revision 3
- RG 8.9, Revision 1, “Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program”
- RG 8.10, Revision 1R
- RG 8.13, Revision 3, “Instruction Concerning Prenatal Radiation Exposure”
- RG 8.15, Revision 1, “Acceptable Programs for Respiratory Protection”
- RG 8.27, , “Radiation Protection Training for Personnel at Light-Water-Cooled Nuclear Power Plants,”
- RG 8.28, “Audible Alarm Dosimeters”

- RG 8.29, Revision 1, “Instruction Concerning Risks from Occupational Radiation Exposure”
- RG 8.34, “Monitoring Criteria and Methods to Calculate Occupational Radiation Doses”
- RG 8.35, “Planned Special Exposures”
- RG 8.36, “Radiation Doses to Embryo/Fetus”
- RG 8.38, Revision 1, “Control of Access to High and Very High Radiation Areas of Nuclear Power Plants”

Design Considerations

The plant radiation protection design should ensure that individual doses and total person roentgen equivalent man (rem) doses to plant workers and to members of the public are ALARA, and individual doses are maintained within the limits of 10 CFR Part 20. FSAR Tier 2, Section 12.1.2 describes design features which serve to minimize the time employees spend in radiation areas and to minimize radiation levels in areas routinely occupied and housing equipment requiring attention by plant personnel. These design features address the following objectives:

- Reduce radiation fields through system design and facility layout, thereby allowing operations, maintenance, and inspection activities to be performed in reduced radiation fields
- Reduce access, repair, and equipment removal times, thereby reducing the time spent in radiation fields
- Attain optimal reliability and maintainability, thereby reducing maintenance requirements for radioactive components
- Minimize to the maximum extent possible the cobalt content of alloys chosen for reactor coolant system (RCS) materials, thereby minimizing the production, distribution, and retention of activated corrosion products throughout the primary system

In addition, FSAR Tier 2, Section 12.1.2 describes several design features which satisfy the objectives of the plant’s radiation protection program:

- The use of highly reliable equipment reduces the frequency of maintenance and associated personnel exposure.
- Except in limited applications where it is necessary for reliability considerations, materials in contact with the RCS have low concentrations of cobalt. This reduces the amounts of cobalt-60 introduced in the RCS. (Cobalt-60 is the primary long term source of radiation fields in nuclear power plants and, therefore, is a major source of radiation exposure during shutdown, maintenance, and inspection activities at light water reactors (LWRs).)
- Adequate spacing and laydown areas facilitate access for maintenance and inspections and allow room for low dose waiting areas, as well as staging of tools and equipment.
- The amount of time spent in radiation areas is minimized with enhanced servicing convenience for anticipated maintenance or potential repairs, including ease of disassembly

and modularization of components for replacement or removal to a lower radiation area for repair or servicing.

- Systems are segregated, such that radioactive systems are separated from nonradioactive systems, and radioactive components are located in separate shielded cubicles. This allows workers to perform maintenance on a given component, while being shielded from the dose of adjacent components.

- Plant design provides anterooms that serve as entries to higher dose rate rooms to protect workers from radiation within the rooms.

- Special tools and remotely controlled equipment are provided for operations that result in significant exposure to personnel, such as change-out of purification filters.

- Equipment and piping are designed to minimize the accumulation of radioactive materials.

- Radioactive systems located within the Nuclear Island avoid field run piping, thereby eliminating a potential source of exposure.

- Flanged connections are provided for large vessels that may require decontamination prior to major maintenance activities. These connections are located outside the vessel room in order to shield the worker and minimize radiation exposure.

- Drains are located at low points.

- Flushing connections are used to minimize the buildup of crud in system components.

- Services such as compressed dry air, demineralized water, and communications equipment are provided in local compartments where experience has shown a need for them.

These design considerations incorporate the basic management philosophy guiding the design effort and conform to the guidelines in RG 8.8. Therefore, the staff finds them acceptable.

In addition to the features described above, the reactor design incorporates several features that represent improvements over many currently operating plants:

- Compartmentalization of components for all major radioactive systems, including the reactor coolant system, chemical and volume control system (CVCS), primary coolant purification system, primary coolant degasification system, fuel pool cooling and purification system (FPCPS), liquid waste management system, gaseous waste processing system, solid waste management system, and the ventilation system

- Installation of permanent platforms and location of equipment such that the need for scaffolding is eliminated

- Clear separation of clean areas from potentially contaminated areas

The design features described in FSAR Tier 2, Section 12.1.2 are intended to minimize personnel exposures and conform to the guidelines of RG 8.8. As such, these design features will help to maintain individual doses and total person-rem doses to plant workers and to members of the public ALARA, while maintaining individual doses within the limits of 10 CFR Part 20. Therefore, the staff finds these design features to be acceptable.

Operational Considerations

Operational considerations regarding the implementation of a radiation protection program are outside the scope of this design certification review. The applicant has stated that a COL applicant who references the certified design will address operational considerations consistent with the level of detail provided in RG 1.206. Section 12 of RG 1.206 lists the following RGs that the COL applicant should address:

- RG 8.2, "Guide for Administrative Practices in Radiation Monitoring"
- RG 8.7, Revision 2, "Instructions for Recording and Reporting Occupational Radiation Exposure Data"
- RG 8.9, Revision 1, "Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program"
- RG 8.13, Revision 3, "Instruction Concerning Prenatal Radiation Exposure"
- RG 8.15, Revision 1, "Acceptable Programs for Respiratory Protection"
- RG 8.27, "Radiation Protection Training for Personnel at Light-Water-Cooled Nuclear Power Plants,"
- RG 8.28, "Audible Alarm Dosimeters"
- RG 8.29, Revision 1, "Instruction Concerning Risks from Occupational Radiation Exposure"
- RG 8.34, "Monitoring Criteria and Methods to Calculate Occupational Radiation Doses"
- RG 8.35, "Planned Special Exposures"
- RG 8.36, "Radiation Doses to Embryo/Fetus"
- RG 8.38, Revision 1, "Control of Access to High and Very High Radiation Areas of Nuclear Power Plants"

Addressing the above RGs is outside the scope of the design certification review. In FSAR Tier 2, Section 12.1.3, "Operational Considerations," the applicant states that COL Information Item 12.1-1 will address operational considerations of the SRP consistent with the level of detail provided in RG 1.206. The applicant also listed RGs the COL applicant will need to address in their application, including RGs as noted above.

12.1.5 Combined License Information Items

The following is a list of COL information items and descriptions from Table 1.8-2 of the FSAR:

Table 12.1-1 U.S. EPR Combined License Information Items

Item No.	Description	FSAR Section	Action Required by COL Applicant	Action Required by COL Holder
12.1-1	A COL applicant that references the U.S. EPR design certification will fully describe, at a functional level, elements of the ALARA program for ensuring that occupational radiation exposures are ALARA. This program will comply with provisions of 10 CFR Part 20 and be consistent with the guidance in RGs 1.8, 8.2, 8.7, 8.8, 8.9, 8.10, 8.13, 8.15, 8.27, 8.28, 8.29, 8.34, 8.35, 8.36, and 8.38, and the applicable portions of NUREG-1736.	12.1.3	Y	

12.1.6 Conclusions

Based on the information supplied by the applicant, as described above, the staff concludes that the U.S. EPR design policy and design features are acceptable. This conclusion is based on the applicant having met the ALARA provisions of 10 CFR 20.1101(b) and the guidance of Regulatory Guides 8.8 (Regulatory Position C.2) and 8.10 (Regulatory Position C.1) with respect to those matters.

The objective of the plant radiation protection design is to maintain individual doses and total person-Sievert (person-rem) doses to plant workers, including construction workers, and to members of the general public ALARA, and to maintain individual doses within the limits of 10 CFR Part 20. The applicant provided training to designers and engineers on incorporating ALARA into the design process. Design changes which affected radiation exposure underwent an ALARA board review with members selected from teams of designers as well as U.S. nuclear power operators.

The applicant incorporated facility and equipment design considerations into the U.S. EPR design to satisfy the radiation protection design objectives listed above. These included the use of low-cobalt alloys, such as Alloy 690, for steam generator tubing, and eliminating to the maximum extent possible the use of cobalt-containing alloys in primary system components. In addition, facility layout compartmentalizes system components and separates nonradioactive systems from radioactive systems. Doses during maintenance are reduced through the use of installed isolation, drain, and vent valves for draining, flushing or decontamination of systems; flanged vessel connections for large vessels are located outside the vessel room to reduce dose to personnel and can be used for decontamination; and components are chosen for high reliability as well as ease of maintenance and replacement. These design features, which are intended to maintain individual doses and total person-rem doses within the limits of 10 CFR Part 20, are in accordance with RG 8.8 and are therefore acceptable.

Operating and maintenance personnel follow specific plans and procedures to ensure that goals related to keeping exposures ALARA are achieved in the operation of the plant. The ALARA operational implementation and policy considerations will be addressed by the COL applicant for the U.S. EPR. The staff finds it acceptable for the applicant to defer discussion of the material addressed by COL Information Item 12.1-1. The staff will determine compliance with the requirements of 10 CFR Part 20 in these areas during the COL review.

12.2 Radiation Sources

12.2.1 Introduction

The determination of projected radiation sources in the plant during normal operations, anticipated operational occurrences, and accident conditions, is used as the basis for designing the radiation protection program and for shield design calculations. This includes definition of isotopic composition, source location in the plant, source strength, and source geometry. In addition, the airborne radioactive material sources in the plant are considered in the design of the ventilation systems and are used for the design of personnel protective measures and for dose assessment.

12.2.2 Summary of Application

FSAR Tier 1: There are no FSAR Tier 1 entries for FSAR Tier 2, Section 12.2 areas of review.

FSAR Tier 2: The applicant has provided an FSAR Tier 2 system description in Section 12.2, which is summarized here in part.

Source terms are presented in Section 12.2 for both the contained and airborne sources of radioactivity that form the basis for shield design calculations and the design of the ventilation systems.

The shielding design-basis primary coolant source term is based on U.S. EPR specific design inputs and a 0.25 percent failed fuel fraction. During normal operation, radiation within the containment consists of neutrons and gamma radiation emitted by the reactor core. This radiation is reduced by shielding provided by the reactor vessel and reactor internals.

Sources of radiation in the RCS are fission products released from fuel cladding defects and activated material in the coolant system. This radioactive material is continuously transported through the large reactor coolant piping. During operation, nitrogen-16 (formed by neutron interaction with nitrogen in exposed water) is the largest source of radioactivity in the reactor coolant and steam generators, and consequently has the most impact on shielding design in the Reactor Building. Because of the seven second half-life of nitrogen-16 and reactor coolant transport times, nitrogen-16 activity varies considerably by location.

During plant operation, radioactive corrosion products deposit on the inner surface of pipes and components. This buildup of contamination is a continuous process, which is mainly dependent on physical and chemical conditions of the RCS in the different states of the reactor (full power, shutdown, and startup). Bounding values of fixed corrosion products for the U.S. EPR are based on operating reactor data for plants with low-cobalt alloys. This information is used to determine shielding requirements during shutdown and for inservice inspection.

The Chemical and Volume Control System (CVCS) processes reactor coolant for purification, degassing, and treatment. The components of this system are located outside of containment. The volume control tank, located in the Fuel Building, is the largest radiological source in the system.

Airborne radioactivity concentrations can occur in the Reactor Building, both during power operation (coolant leakage) and refueling (evaporation of the refueling pool). The spent fuel pool water contains radionuclides from defects in spent fuel and corrosion products from fuel assemblies. The evaporation of the spent fuel pool water can lead to airborne radioactivity concentrations in the Fuel Building, both during power operation and refueling. The airborne radioactivity in the Fuel Building is primarily due to tritium, since the continuous operation of the fuel pool cooling and purification system removes other isotopes from the pool.

Airborne radioactivity concentrations within the Nuclear Auxiliary Building and Radioactive Waste Processing Building result principally from equipment leakage. The ventilation systems in these buildings are designed so that the airflow is from regions of lower potential for contamination to those with higher potential for contamination. As a result, negligible airborne radioactivity concentrations are expected in those areas of the buildings which are normally occupied.

Under normal operating conditions, components within the Turbine Building are expected to contain negligible levels of radioactive material. Airborne radioactivity concentrations in the Turbine Building are, therefore, expected to be negligible.

ITAAC: There are no ITAAC associated with FSAR Tier 2, Chapter 12.

Technical Specifications: There are no technical specifications for FSAR Tier 2, Section 12.2.

12.2.3 Regulatory Basis

1. The relevant requirements of NRC regulations related to projected sources of radiation, and the associated acceptance criteria, are given in Section 12.2 of NUREG-0800 and are summarized below. Review interfaces with other SRP sections can also be found in Section 12.2, “Radiation Sources,” of NUREG-0800.10 CFR 20.1201, “Occupational dose limits for adults,” 10 CFR 20.1202, “Compliance with requirements for summation of external and internal doses,” and 10 CFR 20.1206, ”Planned special exposures,” as they relate to limiting occupational radiation doses.

2. 10 CFR 20.1203 “Determination of external dose from airborne radioactive material” and 10 CFR 20.1204 “Determination of internal exposure,” 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 “Occupational dose limits for minors,” as it relates to limiting exposure to minors to one-tenth of limits for adults.

4. 10 CFR 20.1301 “Dose limits for individual members of the public,” as it relates to the determination of radiation levels and radioactive materials concentrations within the components of waste treatment systems.

5. 10 CFR 20.1801 “Security of stored material,” as it relates to securing licensed materials against unauthorized removal.

6. 10 CFR Part 50, Appendix A, GDC 61, “Fuel Storage and Handling and Radioactivity Control,” as it relates to systems that may contain radioactive materials.

7. 10 CFR 50.34(f)(2)(vii) and GDC 19, “Control Room,” as they relate to the acceptable radiation conditions in the plant under accident conditions, and the source term release assumptions used to calculate estimates of those conditions.

Acceptance criteria adequate to meet the above requirements include:

1. RG 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” as it relates to the assumptions used in evaluating the concentrations of radionuclides in containment and plant systems following a loss-of-coolant accident (LOCA)

2. RG 1.7, “Control of Combustible Gas Concentrations in Containment,” as it relates to methods for determining gaseous concentrations of radionuclides in containment following an accident

3. RG 1.112, “Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors,” as it relates to the calculation of realistic radiation levels and radioactive materials source terms for the evaluation of waste treatment systems

4. NUREG-0737, “Clarification of TMI Action Plan Requirements,” Task Action Plan Item II.B.2, as it relates to the identification of specific post accident sources of radiation in the facility

5. American National Standards Institute/American Nuclear Society (ANSI/ANS) Standard 18.1, “Radioactive Source Term for Normal Operation of Light Water Reactors,” as it relates to the establishment of typical long-term concentrations of principal radionuclides in fluid streams of light-water-cooled nuclear power plants

12.2.4 Technical Evaluation

The staff reviewed the descriptions of the radiation sources given in FSAR Tier 2, Section 12.2, “Radiation Sources,” to assess completeness against the guidelines in RG 1.206 and the criteria in Section 12.2 of the SRP. The applicant will use the contained source terms described in the FSAR as the basis for the radiation design calculations (shielding and equipment qualification) and personnel dose assessment. The applicant will use the airborne radioactive source terms in the FSAR in the design of ventilation systems and for assessing personnel dose. The staff reviewed the source terms in the FSAR to ensure that the applicant had either committed to follow the guidelines of the RGs and staff positions set forth in Section 12.2 of the SRP, or provided acceptable alternatives. Where the FSAR adheres to these RGs and staff positions, the staff can conclude that the relevant requirements of 10 CFR Part 20 and 10 CFR Part 50, Appendix A, GDC 61, have been met.

Contained Sources

In the FSAR Tier 2, Section 12.2.1, “Contained Sources,” the applicant describes the shielding design radiation source terms during normal full-power operation, shutdown, and design basis accident events. Other than the reactor core, the RCS is the principal contributor to radiation levels in the containment. Sources of radiation in the RCS include the following:

- Fission products (which are released from defective fuel cladding)
- Activation products
- Corrosion products

To calculate the fission product shielding design source term, the applicant assumed 0.25 percent fuel cladding defects at full-power operation, except for radioiodines, bromines, and noble gases. The applicant's radioiodines, bromines, and noble gas concentrations were derived from the technical specification dose equivalent (DE) limits for Iodine-131 and Xenon-133. This is an alternate methodology to that given in the guidance of Section 12.2 of the SRP, which calls for a shielding source term based on 0.25 percent failed fuel. The applicant's response to RAI 150, dated January 22, 2009, Question 12.02-1, Part 1 stated that the radioiodines, bromines and noble gas concentrations calculated using the alternate methodology were equal to, or more conservative than, the values obtained using the 0.25 percent failed fuel assumption specified in the guidance of SRP Section 12.2. The staff confirmed that for radioiodines and bromines, the applicant's design-basis concentrations were a factor of 1.67 more conservative relative to concentrations calculated using the 0.25 percent failed fuel assumption. In addition, noble gas concentrations were equivalent using the two methodologies. Therefore, the staff finds that the applicant's methodology is an acceptable alternative to the guidance provided in SRP Section 12.2 and that the U.S. EPR's reactor coolant fission product source term forms a conservative basis for the radiation design calculations and personnel dose assessments needed to demonstrate compliance with the occupational dose limits of 10 CFR Part 20. The staff concludes that RAI 150, Question 12.02-1, Part 1 is resolved.

Additional RCS sources include activation products. Of these, the activation product nitrogen-16 (N-16) is the predominant radionuclide in the RCS piping, reactor coolant pumps (RCPs), and the steam generators (SGs) (all of which are located inside containment) during plant operations. The N-16 activity is not a factor in the radiation sources for systems and components located outside containment during normal power operations because of the short half-life (7.11 seconds) of N-16.

With regard to the last source of radiation in the RCS, design basis corrosion product activity levels, the applicant based their values on operating reactor data for plants using low-cobalt alloys. Therefore, the U.S. EPR's design-basis source term values (values used to determine the shielding thickness) for the major corrosion product nuclides, although of the same order of magnitude, are slightly lower than those of average operating plants.

In accordance with the guidance set forth in Section 12.2 of the SRP, FSAR Tier 2, Section 12.2.1 also describes all large contained sources of radiation which are used as the basis for designing the radiation protection program and completing shield design calculations. These sources include the reactor core; RCS; chemical and volume control system; liquid, gaseous, and solid radwaste systems; and other miscellaneous sources. For these contained sources, the applicant provided either the source strength by energy group or the associated maximum activity levels listed by isotope. The FSAR provides system layouts within rooms or cubicles, as well as information about the type and size of components in these systems. However, the FSAR did not discuss a few significant sources which are a radiation protection program and shielding concern, including spent fuel, residual heat removal (RHR) system during refueling, the movable in-core flux mapping system (i.e., the aeroball system), and the safety injection system after a LOCA. In its response to RAI 150, Question 12.02-3 and RAI 150, Question 12.03-12.04-4,both dated May 13, 2009, the applicant provided bounding gamma ray source strengths for a full core of spent fuel at different times after shutdown; safety injection system at various times post-LOCA; movable in-core detectors as a function of time after shutdown; and the RHR system during refueling. The staff performed confirmatory source term calculations for the U.S. EPR spent fuel and found the applicant's values to be conservative.

The guidance in Section 12.2 of the SRP also states that the applicant should include descriptions of any radiation sources containing byproduct, source, and special nuclear materials. In FSAR Tier 2, Section 12.2.1.13, "Miscellaneous Sources," the applicant stated that the COL applicant will address any byproduct, source, or special nuclear material radiation sources greater than 100 millicuries that may warrant shielding design considerations. The applicant identified this issue as COL Information Item 12.2-1.

The core activity release model for a core melt accident is based on the source term model from RG 1.183. The applicant used the resulting source strengths to calculate post-accident dose rates, as well as worker doses incurred during vital area access/activities following an accident. In the event of core degradation, core cooling can be provided by four separate safety injection trains each located in a separate Safeguards Building adjacent to the Reactor Building. The safety injection pumps take suction from the in-containment refueling water storage tank (IRWST) which is located inside containment, thereby reducing the source term that workers and the public would be exposed to in the event of an accident. The staff confirmed the applicant's use of the RG 1.183 source term model and therefore their compliance with the requirements of 10 CFR 50.34(f)(2)(vii). Therefore, the staff finds the use of this accident source term acceptable.

Based on the information provided in FSAR Tier 2, Section 12.2 and the responses to RAI 150, Question 12.02-3 and RAI 150, Question 12.03-12.04-4, as described above, the staff finds that the applicant's description of the U.S. EPR's large contained source terms forms a conservative basis for the radiation design calculations and personnel dose assessments needed to demonstrate compliance with the occupational dose limits of 10 CFR Part 20, except for the following open item. The applicant did not incorporate the source term information provided in their responses to RAI 150, Question 12.02-3 and RAI 150 Question12.03-12.04-4 into Section 12.2 of their FSAR. Therefore, the staff requested that the applicant revise Section 12.2 of the FSAR to include this source term information. **RAI 280, Question 12.02-5, which is associated with the above request, is being tracked as an open item**.

The FSAR also includes the assumptions that the applicant used in arriving at quantitative values for contained and airborne source terms, based on the relevant requirements of GDC 61 and 10 CFR Part 20.

Airborne Radioactive Material Sources

In FSAR Tier 2, Section 12.2.2, "Airborne Radioactive Material Sources," the applicant described the sources of airborne radioactivity for the reactor design. These include leakage of primary coolant inside containment and activation of naturally occurring argon in the reactor pit, leakage from stored spent fuel assemblies, and evaporative losses from the spent fuel pool and the fuel handling area FSAR Tier 2, Section 12.2 describes the assumptions and parameters used to determine the maximum expected airborne radioactivity concentration levels during normal operations in the Reactor and Fuel Buildings.

The Reactor Building is separated into two compartments: the inner equipment compartment and an outer service compartment. The equipment compartments are inaccessible during normal operations, but the outer, or service, compartments may be accessed for equipment staging purposes prior to an outage. The main source of airborne activity in the equipment compartments of the Reactor Building during operation is leakage of primary coolant and activation of naturally occurring atmospheric argon. Sources of airborne activity in the outer service compartments of the Reactor Building include leakage from the inner compartments. Airborne activity in the equipment areas is reduced using a containment internal filtration subsystem which continuously circulates and filters the air to remove radioactive iodine during normal operation. In addition, the nuclear island vent and drain systems are connected directly to the ventilation system rather than venting to containment spaces. The equipment compartments are inaccessible during normal operations, while during maintenance the containment purge system routes air from areas of lower potential airborne contamination (i.e. service areas) to areas of higher potential contamination (i.e., equipment areas). The staff confirmed that these design features, which reduce or contain the airborne radioactivity present in the Reactor Building, adhere to the guidance of RG 8.8 and are therefore acceptable. The applicant's estimated maximum airborne radioactivity concentrations in the service compartments of the Reactor Building will be at or below the concentrations listed in Appendix B, Table 1, Column 3, to 10 CFR Part 20, and therefore demonstrate the ability of the U.S. EPR's engineering controls (such as ventilation design and compartmentalization of radioactive equipment) to maintain airborne concentrations ALARA in accordance with 10 CFR 20.1101(b).

The sources of airborne activity in the Fuel Building are the spent fuel storage pool and equipment areas. Similar to the Reactor Building, FSAR Tier 2, Section 9.1.2, "New and Spent Fuel Storage," states that the fuel building ventilation system will be used to remove airborne contamination from above the spent fuel pool. Ventilation flow rates are routed from areas of lower potential contamination to areas of higher potential contamination. These design features, which contain and reduce airborne radioactivity concentrations, conform to the guidance of RG 8.8 and are therefore acceptable. The applicant's estimated maximum airborne concentrations in the Fuel Building will be below the concentrations established in Appendix B, Table 1, Column 3, to 10 CFR Part 20, and therefore demonstrate the capability of the U.S. EPR design to maintain airborne concentrations in the fuel building ALARA in accordance with 10 CFR 20.1101(b).

The main source of airborne activity in the Safeguard Buildings (1, 2, 3, and 4) is equipment leakage. The Safeguard Buildings are compartmentalized into radiological controlled areas (containing activity bearing systems such as safety injection and the vent and drain systems) and clean areas which contain instrumentation and other nonradioactive systems. FSAR Tier 2, Section 12.3.1.2, "Safeguard Building," and Section 9.4.5, "Safeguard Building Controlled Area Ventilation," state that the ventilation system serving the Safeguard Buildings is divided into two trains of equipment, with one train serving the potentially contaminated areas which are not normally occupied, and the other train separately serving the clean areas. In addition, FSAR Tier 2, Section 12.3.1.2 also states that areas with a high potential for contamination are maintained at a negative pressure relative to areas with lower potential for contamination, such that airflow prevents the spread of airborne contaminants. A review of FSAR Tier 2, Figures 12.3-60 through 12.3-70 and Figure 9.4.5-1, "Safeguard Buildings Air Supply Subsystem," confirmed the physical separation of the normally occupied areas from those rooms containing activity bearing systems, as well as the use of two separate ventilation trains to serve the potentially contaminated and uncontaminated portions of the Safeguard Buildings. The staff confirmed that these design features, which reduce the spread of airborne radioactivity through containment and physical separation, support the applicant's conclusion that airborne concentrations in the normally occupied portions of the Safeguards Building will be negligible.

The applicant estimates that airborne radioactivity concentrations in the corridors and routine access operating areas within the Nuclear Auxiliary and Radioactive Waste Buildings will be negligible. Just as with the rest of the Nuclear Island, air flow is routed from areas of lower potential contamination (such as stairwells, service corridors, or anterooms) to areas of higher potential contamination, before being exhausted from the plant stack. FSAR Tier 2, Section 9.4.8, "Radioactive Waste Building Ventilation System," states that the radwaste building ventilation system has two separate exhaust air systems – system exhaust air and room exhaust air. System exhaust draws air from those rooms containing radioactive material, including those containing tanks or working areas and machinery. These areas are not normally occupied. The room exhaust system draws air from noncontaminated areas. Pumps and valves for radioactive systems are located in separate compartments that are not normally occupied. Accordingly, the applicant estimates that the resulting airborne concentrations in the normally occupied portions of the Radioactive Waste and Nuclear Auxiliary Buildings will be negligible. The staff confirmed that these design features, which are capable of reducing and containing airborne contamination away from normally occupied areas, adhere to the guidance of RG 8.8, In addition, the staff confirmed that industry operating experience, as documented in NUREG 0713, demonstrates that doses due to airborne radioactivity are a small fraction of reported operating reactor collective dose. Accordingly, the staff finds the applicant's conclusion regarding the airborne radioactivity concentrations in the normally occupied areas of the nuclear auxiliary and radwaste buildings to be acceptable.

The guidance of RG 1.206 states that the applicant should describe in Section 12.2 of the FSAR Tier 2 those airborne radioactive sources in the plant that are considered in the designing of the ventilation systems and in specifying appropriate monitoring systems. This description should include a tabulation of the calculated concentrations of airborne radioactive material, by nuclide, for areas normally occupied by operating personnel. Section 12.2.2 of the FSAR Tier 2 describes the assumptions and parameters used to determine the maximum expected airborne radioactivity concentration levels during normal operations in the Reactor, Fuel, Nuclear Auxiliary, Radioactive Waste and Turbine Buildings. In response to RAI 150 dated May 13, 2009, Question 12.02-2 and RAI 235 dated July 13, 2009, Question 12.02-4, the applicant provided additional information so that the staff could review the applicant's methodology and assumptions for calculating airborne concentrations in the Reactor and Fuel Buildings. In response to RAI 235, Question 12.02-4, the applicant stated that a tritium concentration of 1.0 uCi/g as contained in FSAR Tier 2, Table 11.1-2, "RCS Design Basis Source Term," Column 2, is a realistic concentration for tritium rather than the 4.0 µCi/g listed in Column 1 of FSAR Tier 2, Table 11.1-2. This response is consistent with ANSI Standard 18.1 and industry operating experience regarding tritium concentration levels in the RCS and is, therefore, acceptable. Accordingly, staff considers RAI 235 Question 12.02-4 resolved. Except for the open item that is being tracked as RAI 235, Question 12.02-2, the staff's confirmatory calculations verified that the airborne tritium levels in the Reactor Building would remain below 10 CFR Part 20, Appendix B concentration limits for tritium despite the more realistic 1.0 uCi/g source term. Because the airborne concentrations for tritium would remain below the 10 CFR Part 20 Appendix B concentrations and, therefore, would not likely result in occupational doses in excess of 10 CFR Part 20 occupational dose limits, the staff concludes that RAI 235 , Question 12.02-2 is resolved except for the following open item. The staff has requested that the applicant clarify the assumptions used to calculate airborne radioactivity concentrations.

RAI 280, Question 12.02-6, which is associated with the above requests, is being tracked as an open item.

Except for the open item being tracked under **RAI 280 , Question 12.02-6**, the staff finds that the applicant's description of the airborne radioactive source term and associated engineering controls (such as ventilation design and compartmentalization of radioactive equipment away from normally occupied areas) for the U.S. EPR design constitute an acceptable basis for satisfying the applicable requirements of 10 CFR Part 20, including the dose limits of 10 CFR 20.1201 and 20.1202, and the engineering controls requirements of 10 CFR 20.1701, such that doses to workers due to airborne radioactivity in normally occupied areas will be maintained ALARA in accordance with 10 CFR 20.1101(b).

12.2.5 Combined License Information Items

The following is a list of item numbers and descriptions from FSAR Tier 2, Table 1.8-2:

Table 12.2-1 U.S. EPR Combined License Information Items

Item No.	Description	FSAR Section	Action Required by COL Applicant	Action Required by COL Holder
12.2-1	A COL applicant that references the U.S. EPR design certification will provide site-specific information for required radiation sources containing byproduct, source, and special nuclear material that may warrant shielding design considerations. This site-specific information will include a listing of isotope, quantity, form, and use of all sources in this latter category that exceed 100 millicuries.	12.2.1.13	Y	

12.2.6 Conclusions

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 the assumptions used in arriving at quantitative values for these contained and airborne source terms, based on the guidance of SRP Section 12.2 (NUREG-0800), or justified appropriate alternative methodologies. For post-accident shielding for vital area access, the applicant used the source terms in NUREG-0737 and Regulatory Guide 1.183.

During power operation, the greatest potential for personnel dose is inside the containment from N-16, noble gases, and neutrons. The containment is inaccessible during at-power operation except for the outer service compartments in which maximum dose rates do not exceed 0.025 mSv/hr (2.5 mrem/hr) and 0.001 mSv/hr (0.1 mrem/hr) gamma and neutron, respectively. 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 products are based on operating experience data for plants with low-cobalt alloys. Neutron and prompt gamma source terms are based on reactor core physics calculations and operating experience from reactors of similar design. FSAR Tier 2, Section 11.1, “Source Terms,” contains other parameters used, as well as a complete description of the routine operational source term development.

The main sources of airborne radioactivity are from sources located inside containment and the Fuel Building. Leakage from the equipment compartments constitutes the main airborne sources for the service compartments of the Reactor Building. Leakage from stored spent fuel assemblies and evaporative losses from the spent fuel pool are the main sources of airborne radioactivity in the Fuel Building, while equipment leakage is the main source of airborne concentrations in the Safeguard, Nuclear Auxiliary, and Radioactive Waste Buildings. For the Reactor and Fuel Buildings, the applicant has provided a tabulation of the maximum expected routine radioactive airborne concentrations for normally occupied areas. For the Safeguard, Nuclear Auxiliary, and Radioactive Waste Buildings, airborne radioactivity comes mainly from equipment leakage. Operational experience has shown airborne radioactivity to be a negligible contribution to personnel dose, and the U.S. EPR design ensures that any airborne radioactivity is contained and reduced through the use and design of the nuclear island ventilation system. Features such as the use of pressure gradients to direct air flow and the use of separate ventilation trains to serve contaminated areas in the Safeguard and Radioactive Waste Buildings ensures that airborne radioactivity levels will be negligible in the normally occupied areas of the Safeguard, Nuclear Auxiliary, and Radioactive Waste Buildings.

The staff has reviewed the applicant’s submittal against the requirements of 10 CFR Part 20 as it relates to limits on doses to occupationally exposed persons in restricted areas, and the requirements of GDC 61, as it relates to the information on radiation sources provided by the applicant. Except for the open items tracked under **RAI 280, Question 12.02-5 and RAI 280, Question 12.02-6,** the staff finds, for the reasons set forth above, that the applicant’s description of contained and airborne sources complies with the requirements of 10 CFR Part 20 and GDC 61. The staff also finds it acceptable for the applicant to defer discussion of the material addressed by COL Information Item 12.2-1. The staff will determine compliance with this COL information item during the COL review.

12.3 Radiation Protection Design Features (including Dose Assessment)

This section is written to cover both FSAR Tier 2, Sections 12.3, “Radiation Protection Design Features,” and 12.4, “Dose Assessment,” because, for FSAR Tier 2, Section 12.4, the applicant refers to Section 12.3 while NUREG-0800, Section 12.3-4 is written to cover both sections.

12.3.1 Introduction

This section focuses on radiation protection design features, including the equipment used for assuring that occupational radiation exposures will be as low as is reasonably achievable. Dose rates during normal operation, anticipated operational occurrences, and accident conditions are considered. Radiation zones are defined for various modes of plant operation. Design features to control personnel radiation exposures include the physical layout of equipment, shielding and barriers to high radiation areas, fixed area radiation monitors, and continuous airborne radioactivity monitoring instrumentation, including instrumentation for accident conditions. The estimated annual personnel doses associated with major functions, such as operation, handling of radioactive waste, normal maintenance, special maintenance (e.g., steam generator tube plugging), refueling, and inservice inspection provide a measure of the effectiveness of the proposed design features.

12.3.2 Summary of Application

FSAR Tier 1: Section 2.4.22, “Radiation Monitoring System,” provides information on the radiation monitoring system. FSAR Tier 1, Section 2.9.4, “Sampling Activity Monitoring System,” provides information on the sample activity monitoring system. In addition, design features provided in FSAR Tier 1 which demonstrate compliance with the occupational radiation safety requirements of 10 CFR Part 20 include: Dose significant shield walls located in the Nuclear Island and Radioactive Waste Buildings, the containment high radiation accident monitors, the main control room ventilation accident radiation monitors, and the nuclear island ventilation system.

FSAR Tier 2: The applicant has provided a FSAR Tier 2 system description in Section 12.3, summarized here in part, as follows:

Radiation protection design features include shielding, ventilation, radioactivity monitoring systems, and contamination control. Also presented in this section of the application is a projected annual personnel dose assessment for the U.S. EPR.

The inner compartment of the Containment Building contains the steam generators, reactor coolant pumps, and primary loop piping. The Containment Building outer compartment houses support equipment. Shielding is provided within each room or compartment to shield workers from radiation from adjacent equipment. Plant personnel do not routinely enter the Containment Building during power operations.

A hot workshop is provided for receiving, disassembling, repairing, and machining activated or contaminated components to control the spread of contamination and provide a low dose rate area for servicing. A tool store adjacent to the hot workshop is provided for the control, storage, issuance, and receipt of contaminated tools and equipment. Radioactive piping and associated equipment are isolated and drained for routine maintenance.

Ventilation provisions to protect workers from airborne radioactive material include air pressure gradients from low potential airborne contamination areas to areas of higher potential airborne contamination and then exhaust of the air through filters.

Very high radiation areas in the Reactor Building during normal and refueling operations include the reactor cavity, core internals storage area, instrument lance storage, and fuel transfer pit. The very high radiation areas in the Fuel Building during normal and refueling operations include the fuel transfer pit, spent fuel pool, and cask loading pit. These areas are flooded with water or otherwise inaccessible by workers when very high radiation levels are present.

Radiation zones for each plant area are defined by the dose rate in the areas, taking into account sources within each area as well as contributing dose rate from sources in adjacent areas. Radiation zone categories are described in FSAR Tier 2, Table 12.3-2, “U.S. EPR Radiation Zone Designation.”

The area radiation monitoring instrumentation for use during normal operation and abnormal operating occurrences is provided to measure the radiation levels in specific areas of the plant and to create a continuous record of radiation levels at key locations. The instrumentation also warns of possible equipment malfunctions and leaks in specific areas and furnishes information for radiation surveys. The area radiation monitoring instruments for routine monitoring are powered by the non-1E power supply. Area radiation monitoring equipment used during postulated accidents is powered by the emergency uninterruptible power supply, which is served by a 2-hour battery backup with diesel generators as the auxiliary power to provide continuous indication.

Radiation exposures to facility personnel result primarily from direct gamma radiation from components and equipment containing radioactive material. Experience at operating light water reactors indicates that any dose from airborne radioactivity will not be a significant contribution to the total dose. The applicant estimates a total annual occupational radiation exposure for a unit of 0.5 person-sievert (50 person-rem). This includes the activities of reactor operations and surveillance, routine maintenance, inservice inspection, special maintenance (such as steam generator retubing), waste processing, and refueling.

ITAAC: The ITAAC associated with the radiation monitoring system are provided in FSAR Tier 1, Table 2.4.22-3, “Radiation Monitoring System ITAAC.” The ITAAC associated with the sample activity monitoring system are provided in FSAR Tier 1, Table 2.9.4-3, “Sampling Activity Monitoring System ITAAC,” and ITAAC associated with Nuclear Island and Radioactive Waste Building barriers for radiation shielding are provided in FSAR Tier 1, Table 2.1.1-4, “Nuclear Island ITAAC.”

Technical Specifications: Technical specifications for the control of high radiation areas are addressed in FSAR Tier 2, Chapter 16, Technical Specifications, Section 5.7, “High Radiation Area.” Technical Specifications for post accident monitoring instrumentation are addressed in FSAR Tier 2, Chapter 16, Section 3.3.2, “Post Accident Monitoring (PAM) Instrumentation.”

Conceptual Design: This section of the application contains conceptual design information that is outside the scope of the design certification related to the following systems: Access control facilities, including the personnel decontamination area, portable instrument calibration facility, respiratory facility, equipment decontamination facility, radioactive materials storage area, and facility for dosimetry processing and bioassay.

12.3.3 Regulatory Basis

The relevant requirements of NRC regulations for radiation protection design features (including dose assessment), and the associated acceptance criteria, are given in Section 12.3-4 of NUREG-0800 and are summarized below. Review interfaces with other SRP sections can be found in Section 12.3-4, “Radiation Protection Design Features,” of NUREG-0800.

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 “Use of process or other engineering controls,” and 10 CFR 20.1702 “Use of other controls,” as they relate to design features, ventilation, monitoring, and dose assessment for controlling the intake of radioactive materials

4. 10 CFR 20.1301 and 10 CFR 20.1302 “Compliance with dose limits for individual members of the public,” as they relate to the facility design features that impact the radiation exposure

to a member of the public from non-effluent sources associated with normal operations and anticipated operational occurrences

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

6. 10 CFR 20.1601 “Control of access to high radiation areas,” 10 CFR 20.1602 “Control of access to very high radiation areas,” 10 CFR 20.1901 “Caution signs,” 10 CFR 20.1902 “Posting requirements,” 10 CFR 20.1903 “Exceptions to posting requirements,” and 10 CFR 20.1904 “Labeling containers,” as they relate to the identification of potential sources of radiation exposure and the controls of access to and work within areas of the facility with a high potential for radiation exposure

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

8. 10 CFR Part 50, Appendix A, GDC 19 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 (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)(2)(vii)

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

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

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

Acceptance criteria adequate to meet the above requirements include:

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

2. RG 1.52, “Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants,” as it relates to radiation protection considerations for engineered safety feature atmosphere cleanup systems

3. RG 1.69, “Concrete Radiation Shields and Generic Shield Testing for Nuclear Power Plants,” as it relates to occupational radiation protection shielding structures for nuclear power plants

4. RG 1.97, Revision 4, “Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants,” as it relates to instrumentation for radiation monitoring following an accident

5. RG 1.183, as it relates to evaluating doses to individuals accessing the facility during and following an accident

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

7. RG 8.8, as it relates to radiation protection information to be supplied to personnel

8. RG 8.10, as it relates to the commitment by management and vigilance by the radiation protection manager and staff to maintain occupational radiation exposures as low as reasonably achievable

12.3.4 Technical Evaluation

The staff reviewed the radiation protection design features, dose assessment, and minimization of contamination design considerations contained in FSAR Tier 2, Section 12.3, for adherence to the guidelines in RG 1.206 and the criteria in Section 12.03-12.04 of the SRP. The purpose of this review was to ensure that the applicant had either committed to follow the guidelines of the RGs and applicable staff positions, or offered acceptable alternatives. Where the FSAR adheres to these RGs and staff positions, the staff can conclude that the relevant requirements of 10 CFR Parts 20, 50, and 70 have been met. The following sections present the staff's findings.

12.3.4.1 *Radiation Protection Design Features*

The reactor design incorporates several features to help maintain occupational radiation exposures ALARA in accordance with the guidance in RG 8.8. These include facility design, shielding, ventilation, and area and airborne radiation monitors. These design features are founded on the ALARA design considerations described in FSAR Tier 2, Section 12.1 and discussed in Section 12.1 of this report.

12.3.4.1.1 Facility Design Features

The reactor vessel is located low and in the center of the Reactor Building and is well shielded in order to reduce neutron streaming outside the biological shield. Insulation on the reactor coolant system and the reactor vessel is installed using individually identified pieces that connect together with quick-disconnect clasps, thereby facilitating easy removal and installation and reducing the personnel dose associated with the task.

The RCPs are vertical, single-stage, centrifugal pumps. Each RCP assembly has one common vertical shaft line for the pump and motor with main and auxiliary bearings, one single double thrust bearing, and a flywheel located at the top of the motor shaft. In the event of special maintenance, the U.S. EPR design includes a removable shaft and permanently installed decontamination equipment to reduce occupancy times and, therefore, reduce occupational dose. RCP maintenance is performed in a dedicated decontamination room located in the Fuel Building. This will also reduce personnel exposure by providing a work area away from RCS sources.

In currently operating plants, maintenance and inspections of the SGs are a significant ALARA concern. In response to RAI 164 dated March 11, 2009, Question 12.03-12.04-8, Part 1b, the applicant provided additional detail on SG ALARA design features, including: (1) A description of an improved gasket design for reducing leakage from primary and secondary manways; (2)

Use of improved materials and manufacturing processes aimed at reducing tube failures and associated required maintenance; (3) Reduction in the number of welds and, therefore, the number of required inservice inspections; (4) Improved steam generator blowdown system design resulting in less sludge removal during outages, and (5) Incorporation of steam generator bunker walls capable of blocking radiation from the open SG primary side to workers performing maintenance on the associated reactor coolant pump. The staff confirmed that these design features, which are intended to minimize maintenance and SG source term values and associated exposures, conform to the guidance of RG 8.8 for maintaining occupational radiation exposures ALARA in accordance with 10 CFR 20.1101(b). Accordingly, the staff finds the applicant's response acceptable and concludes that RAI 164 Question 12.03-12.04-8 Part 1b is resolved.

The FSAR states that motor-operated, air-operated, or other remotely actuated valves will be employed where justified by the activity levels and frequency of use, to minimize personnel exposures resulting from valve operations. Full ported valves, which minimize flow obstruction, will be used in systems that will be expected to contain radioactive solids, such as spent resin. Pumps and associated piping are arranged to provide adequate space for access to the pumps for servicing. Pumps in radioactive waste systems will be provided with flanged connections for ease of removal, as well as mechanical seals to reduce servicing times. Filters in radioactive liquid systems will be provided with remote handling systems to minimize personnel exposure and radioactive releases to the environment. Instrument devices are located in low radiation zones away from radiation sources, whenever practicable, while instrumentation and control equipment located in the Reactor Building service compartments have instruments and sensing line connections located to avoid corrosion product and radioactive gas buildup. The heating, ventilation, and air conditioning (HVAC) systems will maintain the airflow direction from areas of lower potential airborne contamination to areas of higher airborne contamination.

In addition to designing equipment to comply with ALARA guidelines, the plant layout is designed to reduce personnel exposures, as follows. The design provides adequate work and laydown space at each inspection and maintenance station. In addition, it provides for installed rigging or structural rigging points to facilitate the removal, transport, or replacement of equipment and use of portable or removable shielding during maintenance and surveillance activities. Adequate illumination and support services (e.g., power, compressed air, water, ventilation, and communications) will be available at work stations. Valves associated with highly radioactive components will be separated from other components and will be located in shielded valve galleries. Valve galleries will be further divided into subcompartments, with walls and shielding entrances such that personnel will only be exposed to the valves and piping associated with one component at a given location. Shielded pipe ducts containing radioactive piping will be located on the opposite side of the room from accessible areas such that pipe penetrations will not penetrate the access corridor. In addition, piping compartments will have labyrinth shielding and, in some cases, shielded doors to eliminate the streaming that a high radiation source emits. To minimize radiation streaming through wall penetrations, the design calls for labyrinths and doors where the potential exists for streaming or scattered radiation, or shielded piping penetrations where radiation streaming cannot be avoided. The staff confirmed that the equipment and layout design features described above conform to the guidelines of RG 8.8 for maintaining occupational radiation exposures ALARA. Accordingly, the staff finds these features acceptable.

The design also incorporates several features to minimize the buildup, transport, and deposition of activated corrosion products in the RCS and auxiliary systems. The FSAR states that the design will eliminate, to the maximum extent possible, the use of cobalt-containing alloys in

RCS components, except in cases in which the use of these materials is necessary for reliability purposes. The majority of the materials exposed to primary reactor coolant will have cobalt impurities of no more than 0.05-weight percent cobalt. The major use of nickel-based alloys in the RCS is in the inconel SG tubes. SG tubing will be made of a material such as Alloy 690 with a limited cobalt content not exceeding 0.05-weight percent. Alloy 690 is in use at currently operating plants and has been shown to result in fewer corrosion products and, therefore, lower dose rates. Similarly, the applicant commits to the use of low cobalt content and corrosion-resistant materials for auxiliary system components (such as valves) which would further reduce corrosion products available for activation. When compatible with the process and purpose, those components which come into contact with reactor coolant, such as the coolant storage tanks, will be made of stainless steel.

The presence of antimony in RCP journal bearings in some current generation plants has increased the number of hot particles at these plants creating an occupational radiation safety concern. In response to RAI 164 dated March 11, 2009, Question 12.03-12.04-8 Part 1a, the applicant stated that the carbon portion of the reactor coolant pump bearings that comes into contact with the reactor coolant will have no antimony. In addition, RCS parts that also come into contact with reactor coolant will have no added antimony, other than trace amounts that will be restricted as described in FSAR Section 5.2.3.1, "Material Specifications." The minimization of antimony in the RCP journal bearings and RCS parts will provide additional reduction in the corrosion products available for activation. This design feature, which minimizes the transport, deposition, and buildup of activated corrosion products in the RCS and auxiliary systems, conforms to the guidelines of RG 8.8 for maintaining occupational radiation exposures ALARA and is, therefore, acceptable. The staff confirmed that Revision 1 of the FSAR, dated May 29, 2009, contains the changes committed to in the RAI response. Accordingly, the staff concludes that RAI 164, Question 12.03-12.04-8, Part 1a, is resolved.

In addition to reducing the source of activated corrosion products through material selection, Section 5.2.3 of the U.S. EPR FSAR states that concentrations of corrosion products will be further reduced through the monitoring and controlling of primary-side water chemistry in accordance with the U.S. EPRI PWR Primary Water Chemistry Guidelines. By inhibiting primary-side corrosion-induced degradation through chemistry control, the applicant will minimize the number of corrosion products that pass through the core and therefore the number of activated corrosion products that can contribute to worker exposure. A high percentage of the remaining corrosion product concentrations in the RCS can be removed through the use of the CVCS filters.

Tanks and piping have also been designed to maintain personnel exposures ALARA. Crud traps created in weld areas will be minimized by the use of butt welds for resin slurry and evaporator bottoms. Tanks containing radioactive liquid will be fabricated from stainless steel, have drain pipes connected at the lowest part of the tank, and will have convex or sloped-bottom designs to minimize radioactivity deposition, unless special features such as agitators, stirrers, or decontamination provisions are provided. Non-removable backing rings in the piping joints will be prohibited, eliminating a potential crud trap for radioactive materials. Exposed surfaces will be designed to minimize contamination. Equipment and piping containing radioactive materials will have provisions for draining and flushing.

In FSAR Tier 2, Figures 12.3-13 through 12.3-71, the applicant provided the staff with detailed drawings of the plant layout which indicate the eight or more radiation zones used in the plant design in addition to shielding wall thicknesses. These radiation zones serve as a basis for classifying occupancy and access restrictions for various areas within the plant during normal operations and accident conditions. Radiation zones were defined according to the dose rate in the area taking into account the sources located in the room as well as those in adjacent rooms. Frequently accessed areas are shielded as Zone 3 (dose rates less than or equal to 0.025 mSv/hr [2.5 mrem/hr]) allowing workers to spend up to a work year (2000 hours at these locations without exceeding the 10 CFR 20.1201 occupational dose limit of 5 rem whole body. FSAR Tier 1, Table 2.1.1-4, "Nuclear Island ITAAC," provides inspections, tests, analysis and acceptance criteria for ensuring (prior to fuel load) the proper thicknesses of walls which serve as significant radiation barriers during normal operation and accidents. Initial plant start up tests described in FSAR Tier 2, Sections 14.2.12.16.1, "Low Power Biological Shield Survey (Test No. 193) and 14.2.12.19.1, "Biological Shield Survey (Test No. 212), provide additional assurance of compliance with 10 CFR Part 20 by requiring that radiation surveys be performed in accessible areas outside the biological shield at 5 percent, 50 percent and greater than or equal to 98 percent power levels. These radiation surveys will verify that radiation zones and associated occupancy times during power operation will be as defined in chapter 12 of the FSAR. On the basis of its review of the ITAAC, the initial plant start up tests, and the detailed zoning drawings, as set forth above, the staff concludes that the applicant's method of plant zoning for normal operations conforms to the guidance in RG 1.206 and the SRP and is, therefore, acceptable.

As required by 10 CFR 50.34(f)(2)(vii), an applicant must fulfill the following requirements:

- Perform radiation and shielding design reviews of spaces around systems that may, as a result of an accident, contain radioactive materials

- Design, as necessary, adequate access to important areas and protection of safety equipment from the radiation environment

Item II.B.2 of NUREG-0737, dated November 1980, provides additional guidance on how these requirements can be met. Item II.B.2 describes source term information that should be used to calculate post-accident radiation levels. Item II.B.2 states that the post-accident plant dose rates should be such that the dose to plant personnel should not exceed 5E-2 sieverts (5 rem) whole body, or its equivalent to any part of the body, for the duration of the accident (per 10 CFR Part 50 and GDC 19). The dose rate in areas that are continuously occupied should be less than 15E-5 sieverts per hour (15 millirem per hour) over 30 days.

Item II.B.2 of NUREG-0737 describes a "vital area" as any area that will, or may, be occupied to permit an operator to aid in the mitigation of, or recovery from, an accident. Item II.B.2 also recommends listing all vital areas in the plant, and providing a summary of the integrated doses to personnel for each of the plant areas that are either continuously occupied or infrequently accessed for the duration of the accident. (These doses should include exposure received while in transit between vital areas.) FSAR Tier 2, Section 12.3.5.2, "Postaccident Access to Radiological Vital Areas," lists all of the vital plant areas that may be accessed postaccident, as well as associated integrated mission doses, and states that all vital areas can be accessed following an accident with exposure less than .05 sieverts (5 rem) to the whole body or .5 sieverts (50 rem) to the extremities.

FSAR Tier 2, Figures 12.3-64 through 12.3-71 contain plant radiation zone maps which reflect maximum radiation fields over the course of an accident. The radiation zones for Division 4 of the Safeguard Building are the same as those for Division 1 due to the symmetrical layout of the U.S. EPR. The applicant performed analyses and confirmed that the individual exposure limits following an accident will not exceed the applicable requirements of GDC 19. The analysis performed assumed the use of full protective clothing and respiratory protection (and therefore negligible dose due to airborne radioactivity), temporary shielding, and restrictions on earliest post-accident access.

Compliance with 10 CFR 50.34(f)(2)(vii) was demonstrated based upon assumptions which included operator actions and, therefore, operational procedures, all of which fall outside the scope of the design certification. Therefore, the staff issued RAI 136, Question 12.03-12.04-2, asking the applicant to justify not incorporating a COL information item to demonstrate compliance with the requirements of 10 CFR 50.34(f)(2)(vii). In a response dated February 25, 2009, the applicant stated that inclusion of COL Information Item 12.5-1 in FSAR Tier 2, Section 12.5, which states that the COL applicant will describe its radiation protection program, would ensure that access times, respiratory protection, and temporary shielding would be used as appropriate during post-accident vital area access. In response, the staff issued supplemental RAI 235, Question 12.03-12.04-11, requesting the applicant to estimate the maximum airborne concentrations present in the radiological vital areas such that the staff could evaluate whether mission dose due to direct radiation from airborne radioactivity would, in fact, be negligible, as stated in the U.S. EPR FSAR Section 12.3. In response, the applicant provided a mark-up of the U.S. EPR FSAR, Tier 2, Section 12.3, which included a table listing estimated maximum airborne concentrations for all radiological vital areas. In addition, the applicant provided a mark-up of U.S. EPR FSAR, Tier 2, Table 12.3-12, "U.S. EPR Estimated Accident Mission Dose," that included the airborne radioactivity contribution to the total mission dose for each vital area. The staff finds the FSAR mark-ups to be acceptable. The staff confirmed that the applicant's analysis approach is based on the guidelines of Item II.B.2 of NUREG 0737 and is, therefore, acceptable. As a result, the staff concludes that RAI 136, Question 12.03-12.04-2 is resolved. **RAI 235, Question 12.03-12.04-11**, is being tracked as a confirmatory item and will be evaluated when the next revision to the U.S. EPR FSAR is submitted to the NRC.

The regulation in 10 CFR 52.47(a)(2) requires that a design certification application must contain a representative conceptual design for those portions of the plant for which the applicant does not seek certification. This information is needed to aid the NRC in its review and to permit assessment of the interface requirements in paragraph 10 CFR 52.47(a)(25). For the FSAR, Chapter 12, conceptual design information is provided for the access control facilities, including the personnel decontamination area, portable instrument calibration facility, respiratory facility, equipment decontamination facility, radioactive materials storage area, and facility for dosimetry processing and bioassay. As described in the guidance of RG 1.206, the COL applicant will include an appropriate site-specific design for these facilities. The staff finds it acceptable for the applicant to defer discussion of the access control facilities to the COL applicant.

Based on the evaluation provided above, the staff finds that the information contained in FSAR Tier 2, Section 12.3.1, "Facility Design Features," and Section 12.3.5.2 adequately addresses the relevant requirements of 10 CFR Part 20, 10 CFR 50.34(f)(2)(vii), and 10 CFR 52.47(a)(24).

12.3.4.1.2 Shielding

The objective of the plant's radiation shielding is to minimize plant personnel and population exposures to radiation during normal operation (including anticipated occupational occurrences (AOOs) and maintenance) and during accident conditions, while maintaining a program of controlled personnel access to and occupancy of radiation areas. The design also includes shielding, where necessary, to mitigate the possibility of radiation damage to materials.

The design physically separates multiple systems into trains. For example, the safety injection trains, which also serve as the RHR trains, are each located in a separate Safeguard Building. Therefore, each train is located in a separate and well-shielded division or building. The physical separation allows for one train to be taken out of service for maintenance while remaining well shielded from the other three trains. In addition, the design provides additional shielding by compartmentalizing the major radioactive components of several systems, including the RCS, chemical and volume control system, primary coolant purification system, fuel pool cooling and purification system, liquid waste management system, gaseous waste processing system, solid waste management system, and ventilation system. Those sources or components which are more radioactive are located on the lower levels of the corresponding building in which they are located.

The FSAR states that radioactive components and piping will be separated from nonradioactive components and piping to minimize personnel exposure during maintenance and inspection activities. Nonradioactive equipment which needs frequent maintenance and inspection is located outside of radiation areas. When radioactive piping is routed through corridors or other low radiation zones, shielded pipe chases are provided, and piping is routed in the opposite side of the building away from normal personnel traffic areas. In addition, the design segregates radioactive components into compartments, enabling workers to perform maintenance on one component while being shielded from adjacent radioactive components. Where applicable, pumps and other support equipment for components that contain radioactive material are separated from the more highly radioactive components by locating them outside the component cubicle in separate shielded cubicles. Shielded compartments have labyrinth entrances or shielded doors to minimize radiation streaming directly through access openings. Penetrations are located to preclude a direct line from the radioactive source to adjacent occupied areas, or they are well shielded. Space is allocated, where needed, for the erection of temporary shielding. Removable shielding is provided for components that may need to be replaced in high-radiation areas, such as the CVCS high pressure coolers. The staff confirmed that these shielding techniques conform to the guidelines contained in RG 8.8 for protecting plant personnel and the public against exposure from various sources of ionizing radiation in the plant. Accordingly, the staff finds these aspects of the shielding design acceptable.

At currently operating reactors, potentially lethal exposures can occur in the vicinity of the fuel transfer tube when a spent fuel assembly passes through this tube during refueling operations. In response to RAI 150, dated May 13, 2009, Question 12.03-12.04-6, the applicant demonstrated that there were two portions of the spent fuel transfer tube for the design which are accessible to personnel. One location is inside containment, while the other location is inside the Reactor Building Annulus. Access to the portion of the transfer tube located inside containment is controlled via a locked, alarmed door equipped with video surveillance NRC staff's independent calculations confirm the applicant's results and show that dose rates outside the locked door are a maximum of 0.0003 mSv/hr (0.03 mrem/hr) while the spent fuel is in transit (as compared to the applicant's 0.0002 mrem/hr (0.02 mrem/hr) calculated dose rate at this location). However, there is also a portion of the spent fuel transfer tube located inside the Reactor Building Annulus which is accessible via a set of stairs and a concrete labyrinth shielding. This area has no apparent barriers to entry (such as a locked door). In response to RAI **254, dated September 10, 2009, Question 12.03-12.04-14** the applicant provided drawings showing the location of the locked gates and doors that will prevent access to both the unshielded portions of the spent fuel transfer tube during refueling. The applicant also provided an FSAR mark-up stating that gates, access doors, double locks, local and remote alarms and video surveillance will be used to ensure that access to the spent fuel transfer tube is restricted in accordance with the requirements of 10 CFR 20.1602. These design features are in accordance with the guidance of RG 8.38 and are, therefore, acceptable for demonstrating compliance with 10 CFR 20.1602. In addition, in response to **RAI 254 , Question 12.03-12.04-13** the applicant provided an FSAR mark-up of Section 12.3.1.8.1, "Reactor Building," which described the unshielded spent fuel transfer tube areas (inside containment and inside the annulus) described above as potential Very High Radiation Areas during fuel transfer. The staff finds the FSAR mark-up to be acceptable**. RAI 254, Question 12.03-12.04-13 and Question 12.03-12.04-14** are being tracked as confirmatory items and will be evaluated when the next revision to the U.S. EPR FSAR is submitted to the NRC.

Unplanned exposures have also occurred in the vicinity of movable in-core detectors that have become stuck during transit outside of the reactor vessel. These stuck in-core detectors have the ability to create high or very high radiation areas and, therefore, pose an occupational radiation safety concern. However, in contrast to the currently operating U.S. reactor fleet, the U.S. EPR design uses an electromechanical computer-controlled, fully automated, online flux mapping measurement system called the Aeroball Measurement System (AMS). In response to RAI 150 dated May 13, 2009, Question 12.03-12.04-4, the applicant provided greater detail on the operation and design of the system as well as the associated source term and radiological hazard that this system could be expected to pose. The applicant stated that the AMS is an online flux mapping system based on movable activation probes, or stacks of 2,500, 1.7 mm (0.07 inches) diameter steel "Aeroballs," composed of carbon, chromium, iron, and vanadium. These Aeroballs are moved into and out of the core pneumatically using high purity nitrogen. The applicant stated that after use, the main isotopic constituents of the Aeroballs - Vanadium-52 and Manganese-56 – would decay away after 45 minutes and 24 hours, respectively and, therefore, would pose no radiological hazard to personnel that may need to access the system 24 hours after use. The staff confirmed that after 24 hours more than 99.8 percent of the Vanadium-52 and Manganese-56 will have decayed away. In addition, because the AMS measurement room is located inside the containment biological shield, it would not be accessible while at-power, and access during outages would be prohibited before the 24 hour decay period. The applicant's response to Question 12.03-12.04-4 stated that to provide additional protection, the AMS equipment area and measuring room will be designated as High Radiation Areas and will be locked to prevent entry. When a measurement is performed, an automatic alarm will sound one minute before the Aeroball stacks enter the measuring room and the AMS measuring room will be automatically locked, thereby preventing personnel entry. However, should personnel be located in the room for some reason, a crash bar will permit personnel inside of the measuring room an unobstructed exit. The applicant also provided relevant AMS operating experience from 12 European and 1 South American plant with 20 - 30 years of system operation each. Over this time period, only three abnormal operational occurrences have occurred and none of these included personnel radiation exposures. The staff confirmed that the information provided from the applicant regarding the source term, operation, and access restrictions associated with the AMS conforms with the guidance of RG 8.8 and RG 8.38 for protecting plant personnel and the public against exposure from various sources of ionizing radiation in the plant. Accordingly, the staff finds these design features acceptable.

Generic Safety Issue (GSI) 137, "Refueling Cavity Seal Failure," and NRC Bulletin 84-03, "Refueling Cavity Water Seal," called for reactor licensees to address the potential for inadvertent reactor cavity drain down via the cavity water seal as well as the associated potential for uncovering spent fuel, either stored or in transit. In accordance with a resolution proposed for GSI 137 as documented in NUREG-0933, the U.S. EPR FSAR Tier 2, Section 3.8.3.11, "Reactor Vessel Support Structure and Reactor Cavity," describes the use of a permanent reactor cavity seal. However, the applicant does not address how the U.S. EPR design or operational procedures will prevent reactor cavity drain down via the reactor cavity access hatch located on the refueling cavity floor, or via reactor cavity drain pipes. The applicant also does not describe a safe laydown area for fuel in transit. GDC 61 requires that the fuel storage and transfer system, in addition to any other system which may contain radioactivity, be designed to ensure adequate safety and shielding, as well as designed to prevent the release of radioactive material during normal and accident conditions.

In addition, the staff noted that the U.S. EPR has an access hatch which opens directly onto the refueling cavity floor. The intent of the hatch is to facilitate worker access to the reactor vessel head. FSAR Tier 2, Section 12.3.1.8.1, “Reactor Building,” states that this access room is equipped with double doors to prevent workers from entering the reactor cavity. The staff requested that the applicant provide more information on the dose rates inside and directly outside this access room while fuel is being moved. If the area is designated by the applicant as a high or very high radiation area during fuel movement, they were also asked to describe the associated design access controls to demonstrate compliance with the requirements of 10 CFR 20.1601 or 10 CFR 20.1602. In addition to a possible high or very high radiation area during refueling, the access hatch described above also provides a potential point of inadvertent reactor cavity drainage during refueling as discussed in GSI 137. The applicant was, therefore, requested to demonstrate compliance with GDC 61, by providing information on the design features associated with the access hatch and reactor cavity, such that the potential for loss of reactor cavity water inventory would be minimized and personnel exposures due to spent fuel would be maintained below the occupational dose limits of 10 CFR Part 20. **RAI 280, Question 12.03-12.04-17, which is associated with the above request, is being tracked as an open item**.

The guidance in Section 12.3.2 of the SRP states that the applicant should describe how the shielding parameters were determined, including pertinent codes, assumptions, and techniques used in the shielding calculations. The FSAR described the shielding codes used to determine the adequacy of the station shielding design. Specifically, the applicant stated that it used the point kernel shielding code MicroShield 7 to calculate most gamma dose rates throughout the plant and the Monte Carlo N-Particle (MCNP) Version 4c code to determine neutron and gamma dose rates inside the Reactor Building. However, for post-accident conditions such as those present in the radiological vital areas that must be accessed post-LOCA, the applicant used the RANKERN Code Version 15a (i.e., a point-kernel program developed by British Nuclear Fuels, plc for gamma ray transport solutions). However, RANKERN is not a shielding code commonly used in the U.S.; therefore, little is known regarding the conservatism of the underlying radiation transport assumptions. Therefore, the staff has requested that the applicant provide the staff with the post-LOCA source term geometries and locations used as input to the RANKERN code, **RAI 254, Question 12.03-12.04-15, which is associated with the above requests, is being tracked as an open item**. In addition, the staff has requested additional information on the Quality Assurance check performed by AREVA on those codes, such as RANKERN, which are not commonly used in the U.S. **RAI 296, Question 12.03-12.04-19, which is associated with the above request, is being tracked as an open item**. The staff determined that the applicant’s assumptions regarding source terms, cross sections, radiation transport, and shield and source geometries are appropriate to the reactor design and conservative. Therefore. the staff finds that the use of MicroShield 7 and MCNP Version 4c shielding codes to be acceptable to evaluate the adequacy of the station shielding design. The staff will not make a finding regarding the RANKERN Version 15A shielding code until the open items being tracked under RAI 254, Question 12.03-12.04-15 and RAI 296, Question 12.3-12.4-19 are resolved and evaluated by the staff.

In addition to shielding walls, RG 8.8 has criteria for penetrations for shielding walls for the purpose of ensuring that workers do not exceed 10 CFR 20.1201 dose limits due to streaming from inadequately placed and/or shielded holes in walls that serve as radiation barriers during normal operations or post accident radiological vital area access. For example, during post-LOCA vital area missions, operators may need to operate valves located in corridors outside of the equipment rooms. The applicant's response to **RAI 254, dated September 10, 2009, Question 12.03-12.04-12** described how penetrations for these valves would be pressure tight and shielded such that local dose rates would be maintained less than 100 mrem/hour. These design features conform to the applicable guidance of RG 8.8 and are, therefore, acceptable for maintaining exposures ALARA in compliance with 10 CFR Part 20. The applicant also provided an FSAR mark-up which revised Section 12.3.5.2, "Postaccident Access to Radiological Vital Areas," to include the criteria that penetrations between the post-LOCA mechanical rooms and the corridors where operators would be located would be pressure tight and shielded. The staff finds the FSAR mark-up acceptable. **RAI 254, Question 12.03-12.04-12** is being tracked as a confirmatory item and will be evaluated when the next revision of the U.S. EPR FSAR is submitted to the NRC.

Except for the issues identified in **Confirmatory Items 254, Questions 12.03-12.04-11, 12.03-12.-4-12, 12.03-12.04-13 and 12.03-12.04-14, RAI 254 , Questions 12.03-12.04-15; RAI 280 , Question 12.03-12.04-17 and RAI 296, Question 12.3-12.4-19** the staff finds that, for the reasons set forth above, the information contained in FSAR Tier 2, Section 12.3.2, "Shielding," adequately addresses the relevant requirements of 10 CFR Part 20, 10 CFR 50.34(f)(2)(vii), and GDC 61.

12.3.4.1.3 Ventilation

RG 8.8 contains guidance on ventilation design features acceptable to control airborne radioactivity levels and maintain personnel doses ALARA. The ventilation systems are designed to protect personnel and equipment from extreme environmental conditions, and to ensure that personnel exposure to airborne radioactivity levels is minimized and maintained ALARA and within the applicable limits of 10 CFR Part 20. Further, the design ensures that the dose to control room personnel during accident conditions will not exceed the limits specified in GDC 19.

The source of airborne radioactivity for a room or area is primarily from equipment leakage within the specified area. The design incorporates the following features to minimize this leakage and thereby reduce the sources of airborne radioactivity:

- Ventilation air is supplied directly to the clean areas of the plant and exhausted from the potentially contaminated areas, thereby creating a positive flow of air from clean areas to potentially contaminated areas.

- Negative or positive pressure is used appropriately in plant areas to prevent exfiltration or infiltration of possible airborne radioactive contamination, respectively.

- Equipment vents and drains are piped directly to a collection system, preventing contaminated fluid from flowing across the floor to a drain and creating a potential airborne contamination problem.

The requirements of 10 CFR Part 50, Appendix A (GDC 19) state that adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposure in excess of .05 sieverts (5 rem) whole body, or its equivalent to any part of the body, for the duration of the accident. In FSAR Tier 2, Section 12.3.5.2, “Post Accident Access to Radiological Vital Areas,” The applicant included the main control room as a vital area. As discussed in Section 12.3.4.1 of this report, the applicant has performed analyses to ensure that individual exposure limits following an accident in vital areas will not exceed the applicable requirements of GDC 19.

These ventilation design features, which are designed to protect personnel and equipment from extreme environmental conditions, and to ensure that personnel exposure to airborne radioactivity levels is minimized and maintained ALARA, conform to the guidance of RG 8.8 and are, therefore, acceptable. Further, the main control room ventilation system is designed to provide controlled overpressure as well as recirculation and filtration of room air, helping to ensure that dose to control room personnel during accident conditions will not exceed the limits specified in GDC 19. Accordingly, the staff finds the ventilation systems design comply with the requirements of 10 CFR Part 20 and GDC 19 and are acceptable.

12.3.4.1.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

The area radiation and airborne radioactivity monitors are discussed in FSAR Tier 2, Section 12.3 and Section 11.5, “Process and Effluent Radiological Monitoring and Sampling Systems.”

The plant area radiation monitoring equipment alerts operators and other station personnel to changing or abnormally high radiation conditions in the plant to prevent possible personnel overexposures and aid health physics personnel in keeping worker doses ALARA. The area radiation monitors supplement the personnel and area radiation survey provisions of the health physics program, which is described in FSAR Tier 2, Section 12.5, “Radiation Protection Operational Program.” The area radiation monitors must comply with the applicable requirements of 10 CFR Part 20, 10 CFR Part 50, and 10 CFR Part 70, and should conform to the personnel radiation protection guidelines of RG 1.97, RG 8.2, and RG 8.8.

Control room displays provide information on monitor readings, alarm set points, and operating status. The area radiation monitors are located according to the potential for significant radiation levels in an area and the expected occupancy of the area. Specifically, area monitors will be installed in the following locations:

- Areas that are normally accessible and where changes in normal plant operating conditions can cause significant increases in exposure rates above those normally designated for the areas

- Areas that are normally or occasionally accessible where significant increases in exposure rates might occur because of operational transients or maintenance activities

In order to inform personnel of local dose rates in the area, area radiation monitors include a local readout and audible alarm in addition to readouts and alarms in the main control room. In addition, visible alarms are located outside each monitored area, so that operating personnel can see them before entering the monitored area. Section 12.03-12.04 of the SRP references ANSI/ANS Standard HPSSC-6.8.1-1981, “Location and Design Criteria for Area Radiation Monitoring Systems for Light-Water Nuclear Reactors,” dated May 1981, which provides acceptable guidance on the location and design criteria of area radiation monitoring systems including recommendations for the installation of monitors in the reactor, auxiliary, and radwaste buildings. However, the applicant did not locate area monitors in the Radioactive Waste Building nor in the primary sampling, laboratory or hot workshop areas in accordance with the guidance set forth in ANSI/ANS HPSSC-6.8.1-1981. In response to RAI 150 dated January 22, 2009, Question 12.03-12.04-5, the applicant incorporated additional area radiation monitors into its design to address this issue, including the drumming room and decontamination room in the Radioactive Waste Building; and the primary sampling, active laboratory, and hot workshop rooms in the Nuclear Auxiliary Building. Because the location and design criteria of these and other area radiation monitors (including the use of local and remote monitor readouts and alarms) conforms to the criteria of ANSI/ANS Standard HPSSC-6.8.1-1981 and the guidance of RG 8.8, the staff finds the response acceptable. The staff confirmed that Revision 1 of the FSAR, dated May 29, 2009, contains the changes committed to in the RAI response. Therefore, the staff concludes that RAI 150, Question 12.03-12.04-5 is resolved.

However, the applicant did not discuss the calibration of the area and airborne monitors other than to state that calibration would be performed in compliance with 10 CFR 20.1501. Therefore, staff requested that the applicant provide information on the calibration methodology and frequency for installed area radiation monitors. Calibration of installed airborne radiation monitors is addressed in Section 11.5 of this report**. RAI 295, Question 12.03-12.04-18, which is associated with the above request, is being tracked as an open item.**

The requirements of 10 CFR 70.24 specify the use of a monitoring system capable of detecting a criticality in designated areas where specified quantities of special nuclear material are handled, used, or stored. In lieu of installing a criticality monitoring system, the applicant has chosen to meet the design and analysis requirements specified in 10 CFR 50.68(b) to demonstrate the prevention of criticality. Performance of a 10 CFR 50.68(b) analysis is an acceptable alternative to compliance with 10 CFR 70.24. . Additional detail regarding the acceptability of the applicant's 10 CFR 50.68(b) analysis is provided in Section 9.1.1, "Criticality Safety of New and Spent Fuel Storage and Handling," of this report.

The requirements of 10 CFR 50.34(f)(2)(xvii) (corresponding to Item II.F.1(3) of NUREG-0737) specify, in part, that the control room must include instrumentation to measure, record, and read out containment radiation intensity (high level). Further guidance is provided in Item II.F.1(3) of NUREG-0737, which indicates that the reactor containment should be equipped with two physically separate radiation monitoring systems that are capable of measuring up to 105 grays (Gy) per hour (107 roentgen per hour) in the containment following an accident. In FSAR Tier 2, Section 12.3.4.1.3, "In-containment High-Range Monitoring," the applicant stated that the design incorporates four electrically independent ion chambers located inside the containment to measure high range gamma radiation. These detectors will be mounted on the inner containment wall in widely separated locations, and will have an unobstructed "view" of a representative volume of the containment atmosphere. The staff confirmed that the design and qualification of these monitors conforms to the guidelines of RG 1.97; SRP Branch Technical Position (BTP) 7-10, "Guidance on Application of Regulatory Guide 1.97;" and NUREG-0737, Item II.F.1(3), with respect to detector range, response, redundancy, separation, onsite calibration, and environmental qualification. In addition FSAR Tier 1 ITAAC Table 2.4.22-3, "Radiation Monitoring System Inspections, Test, Analyses and Acceptance Criteria," provides assurance that the function, environmental qualification, seismic qualification, power supply and alarms for these safety related monitors will be inspected and verified prior to fuel load. Based on the above information provided in Tier 1 and Tier 2 of the FSAR, the staff finds the monitors to be acceptable for demonstrating compliance with the requirements of 10 CFR 50.34(f)(2)(xvii).

The airborne radiation monitoring equipment will be placed in selected areas and ventilation systems to give plant operating personnel continuous information about the airborne radioactivity levels throughout the plant. The airborne radioactivity monitors are located upstream of the filter trains to monitor representative radioactivity concentrations from the areas being sampled. The guidance in Section 12.3 of the SRP indicates that airborne radioactivity monitors should be able to detect the time integrated change of the most limiting particulate and iodine species equivalent to those concentrations specified in Appendix B of 10 CFR Part 20 (one derived air concentration (DAC)) in each monitored plant area within 10 hours (i.e., monitors should be sensitive enough to measure 10 DAC-hours). FSAR Tier 2, Section 12.3.4.2, “Airborne Radioactivity Instrumentation,” and Section 11.5, state that those airborne radioactivity monitors which monitor plant areas which may be occupied by plant personnel will be capable of detecting 10 DAC-hours. Because the applicant conforms to the guidance in Section 12.3 of the SRP, the staff finds the above airborne monitor design description to be acceptable.

The guidance in Section 12.3 of the SRP is that the FSAR will provide the criteria and methods for obtaining representative in-plant airborne radioactivity concentrations in all work areas. The applicant has stated, in FSAR Tier 2, Section 12.3.4.5, “Implementation of Regulatory Guidance,” that the COL applicant will provide site-specific information on the extent to which the guidance provided by RG 1.21, “Measuring, Evaluating, and Reporting Radioactive Material in Liquid and Gaseous Effluents and Solid Waste,” 1.97, 8.2, 8.8, and ANSI/HPS-N13.1-1999 will be employed in sampling, recording, and reporting airborne releases of radioactivity. The applicant has identified this issue as COL Information Item 12.3-1. Further, Item III.D.3.3 of NUREG-0737 (corresponding 10 CFR 50.34(f)(2)(xxvii)) guidance states that each applicant should provide equipment and associated training and procedures for accurately determining the airborne iodine concentrations in areas within the facility where personnel may be present during an accident. The applicant has stated in FSAR Tier 2, Section 12.3.4.5, that the COL applicant will address the use of portable instruments, and the associated training and procedures, to accurately determine the airborne iodine concentration within the facility where plant personnel may be present during an accident, in accordance with the requirements of 10 CFR 50.34(f)(2)(xxvii) and the criteria in Item III.D.3.3 of NUREG-0737. The applicant has identified this issue as COL Information Item 12.3-3. The above considerations concerning representative sampling of airborne concentrations during normal and accident conditions are operational in nature and, therefore, outside the scope of this review. Accordingly, the staff finds it acceptable for the applicant to defer discussion of the material addressed by COL Information Items 12.3-1 and 12.3-3.

12.3.4.2 *Dose Assessment*

The staff reviewed the applicant’s dose assessment contained in FSAR Tier 2, Section 12.3.5, “Dose Assessment,” for completeness against the guidelines in RG 1.206 and the criteria set forth in Section 12.03-12.04 of the SRP. The staff ensured that the applicant had either committed to follow the guidance of the applicable RGs and staff positions set forth in Section 12.3-4 of NUREG-0800, or provided acceptable alternatives. Where the FSAR adheres to these RGs and staff positions, the staff can conclude that the relevant requirements of 10 CFR Part 20 have been met. In addition, the staff selectively compared the applicant’s dose assessment, for specific functions and activities, against the experience of operating pressurized water reactors (PWR). (Radiation exposures to operating personnel shall not exceed the occupational dose limits specified in 10 CFR 30.1201.)

In FSAR Tier 2, Section 12.3.5, the applicant provided an assessment of the annual occupational radiation dose that would be received by the operating staff of a facility. FSAR Tier 2, Tables 12.3-5 through 12.3-11 provide estimated doses for various jobs and inspections that would be performed in the plant during maintenance and refueling periods, as well as for power operations. These activities result in an estimated total annual dose of 0.50 person-sievert (50 person-rem). FSAR Tier 2, Section 12.4 does not contain a separate determination of doses attributable to airborne activity; however, experience at operating light water reactors (LWR) demonstrates that the doses from airborne radioactivity is not a significant contribution to the total dose.

In performing the dose assessment, the applicant reviewed exposure data from similar operating plants in Europe, as well as U.S. operating plants to obtain a breakdown of the doses incurred within each dose assessment category referenced in RG 8.19, "Occupational Radiation Dose Assessment in Light-Water Reactor Plants – Design State Man-Rem Estimates," Revision 1, dated June 1979. The applicant then adjusted these values to account for design features. Based on its calculations, the applicant obtained an estimated annual dose of 0.50 person-sievert (50 person-rem).

The cumulative annual dose of 0.50 person-sievert (50 person-rem) for operating a plant is consistent with the Electric Power Research Institute design guideline of 1.0 person-sievert (100 person-rem) per year and compares favorably with current PWR experience (the 2007 average collective dose for U.S. PWRs was 0.69 person-sievert (69 person-rem)).

As discussed above, the design incorporates several improvements over current operating PWR designs. These improvements are intended to significantly reduce the personnel exposure associated with operational and maintenance activities. The occupational radiation exposure resulting from unscheduled repairs on valves, pumps, and other components will be lower for the U.S. EPR than for current plant designs because of the reduced radiation fields and increased equipment reliability. Historically, special maintenance performed on SGs has resulted in significant personnel doses. The applicant estimates that the annual dose incurred for special maintenance of the SGs will be approximately 0.03 person-Sievert (3 person-rem). These low estimated SG doses result from improved SG design and improved primary and secondary water chemistry controls.

The radwaste system design incorporates a number of dose reducing features including compartmentalization of all major radioactive tanks and components, remote handling equipment, and hydraulic transfer of spent resins to minimize high dose maintenance on plugged piping and pumps. These design features conform to the guidance of RG 8.8 and are therefore acceptable.

The direct radiation at the site boundary of 0.5 miles from the containment and other plant buildings is less than 0.01 mSv (1 mrem) a year and does not exceed 0.02 mSv (2 mrem) in any one hour. The Safeguard and Fuel Buildings have been shielded such that the dose rate immediately outside is less than 0.01 mSv (1 mrem) an hour. This is partly due to the design providing storage for refueling water inside the containment, instead of an outside storage tank, thereby eliminating the refueling water storage tank as an offsite radiation source.

RG 1.206, Section C.I.12.3.5, "Dose Assessment," guidance is that all applicants with multiunit sites should provide estimated annual doses to construction workers due to onsite radiation sources from existing operating plant(s). The applicant has stated in FSAR Tier 2, Section 12.3.5.1, "Overall Plant Doses," that the COL applicant will provide site-specific information, including bases, models, assumptions, and input parameters, on estimated annual doses to construction workers in a new unit construction area as a result of radiation from onsite radiation sources from the existing operating plant(s). This issue is identified by the applicant as COL Information Item 12.3-2 in the FSAR. Dose to construction workers due to new construction is a construction and operational concern and is, therefore, outside the scope of this review. Accordingly, the staff finds it acceptable for the applicant to defer discussion of the material addressed by COL Information Item 12.3-2.

12.3.4.3 *Minimization of Contamination*

The requirements in 10 CFR 20.1406 state that each licensee shall describe how they intend to minimize, to the extent practicable, contamination of the facility and of the environment, as well as the generation of radioactive waste. Applicants are also required to describe how they will facilitate decommissioning. The guidance in Section 12.3-4 of the SRP states that design features described by the applicant should facilitate eventual decommissioning and minimize, to the extent practicable, contamination of the facility and the environment, as well as the generation of radioactive waste. RG 4.21, “Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning,” contains a basis acceptable to the staff for complying with the requirements of 10 CFR 20.1406. Where the applicant adhered to this guidance, the staff can have reasonable assurance of compliance with 10 CFR 20.1406.

FSAR Tier 2, Section 12.3.6, “Minimization of Contamination,” describes a design philosophy of prevention and early detection of leaks, such that occupational doses are maintained ALARA, contamination is minimized and decommissioning is facilitated. The general design features described by the applicant are in accordance with this design philosophy and demonstrate compliance with the requirements of 10 CFR 20.1406. As more fully described below, these features include measures to minimize facility contamination and contamination of the environment, and features to facilitate decommissioning.

The Nuclear Island, consisting of the Reactor Building, Safeguard Buildings 1, 2, 3, and 4, and the Fuel Building, has several design features which demonstrate compliance with the requirements of 10 CFR 20.1406. The Nuclear Island minimizes facility contamination through compartmentalization and containment of spills. Equipment and components subject to leakage are located on the lowest levels of buildings, although still above grade, in order to contain contamination. Equipment such as tanks and pumps are mounted on pedestals to facilitate leakage detection. Tanks are elevated above floor level such that they can be inspected and repaired as necessary in the event of a leak. Tank sampling stations are designed to minimize leakage to the floor and include leakage collection capability in the event of a leak. Berms and collection and drip pans are provided to limit contamination to smaller areas. Tanks are instrumented with both local and remote level indications, and alarms and tank vents are designed to transfer any overflow to receiving tanks. Level instrumentation and other leak detection measures provide a signal to automatically isolate the system or provide indication to the main control room (MCR) to initiate operator action from within the MCR or locally. Coolant storage tank compartments are designed with a leak retention capability equivalent to the complete drainage of one tank. Liquid leak offs that potentially have not been degassed are transferred to tanks that are purged to the gaseous waste processing systems. Floor and equipment drains, as well as sumps that transport contaminated liquids from the Reactor, Safeguard, and Fuel Buildings to the Radioactive Waste Processing Building. Embedded floor drain piping located at the lowest elevations is fitted with a concentric guard pipe and an alarmed moisture detection monitor. Sumps located on the lowest elevation of each building are double lined with non-porous material and are also fitted with alarmed leakage detection instrumentation. In addition, no access openings or tunnels penetrate the exterior walls of the Nuclear Island below grade.

In response to RAI 23 dated October 29, 2008, Question 12.03-12.04-1, the applicant stated that eyewash stations and shower wastewater in the Access Building are routed to a tank in the nuclear island drain and vent system. Liquid effluents generated in the decontamination facilities (such as showers, floor washing) are collected and stored in the storage tanks of the nuclear island drain and vent system. However, the applicant did not incorporate applicable parts of its response to RAI 23, Question 12.03-12.04-1 into the FSAR for the nuclear island drain and vent system and for other systems which contribute to contamination of the facility and the environment. Therefore, the staff requested that the applicant include additional design features in the FSAR. **RAI 228, Question 12.03-12.04-9, which is associated with the above requests, is being tracked as an open item**.

The radioactive waste building provides design features to control and collect radioactive material spills in accordance with 10 CFR 20.1406. Radioactive waste tanks are equipped with level measurements and overflows to prevent uncontrolled overflow paths to the environment. Such tanks are located in rooms designed to either contain spills (using walls and curbs) or drain the spills to sumps equipped with leakage detection systems. These detection systems provide a signal to automatically isolate the affected system or provide an indication to the MCR to initiate operator action either from within the MCR or locally. None of these rooms have doors leading to the outside environment. Sumps are pumped into a storage tank in the liquid waste storage system. Piping and equipment for these systems are stainless steel to avoid corrosion, and there is no embedded piping.

In current operating reactors, the spent fuel pool (SFP) has been the source of leaks that have resulted in low levels of facility contamination. The SFP has several design features that address this potential problem. For example, the SFP is located above the lowest elevation of the Spent Fuel Building and is equipped with a stainless steel liner. There are no spent fuel pool systems that are buried or routed through exterior boundaries. The leakage detection system under the spent fuel pool provides coverage in case of a leak, and leak detection equipment in channels aid in identifying the location of the leak. Sumps that have the potential to collect spent fuel pool leakage are doubled-lined with non-porous material. In addition, walls and curbs are used around locations of potential leaks of contaminated fluids.

Operating experience has also shown that effluent discharge piping can be a source of low level environmental contamination. The applicant has addressed this concern by designing the effluent discharge piping located outside of the Radioactive Waste Processing Building as a concentric guard pipe with an alarmed moisture detection monitor fitted to the outer pipe to detect any possible leaks. The double pipe system extends all the way to the discharge pipe outlet at the cooling water outfall. In addition, the secondary coolant treatment waste and turbine building drain systems are designed with grab sample provisions, while the clean drain system (which encompasses the non-contaminated waste water, as well as the turbine building clean drain system) has a radiation monitor on the common discharge line along with grab sample provisions. These design features for the effluent discharge piping, as well as for the secondary coolant treatment waste, turbine building drain, and clean drain systems comform to the guidance in RG 4.21, and are therefore acceptable.

In response to RAI 23 dated October 29, 2008, Question 12.03-12.04-1, the applicant described design features for the coolant storage and transfer system which demonstrated compliance with 10 CFR 20.1406, including the fact that it was isolated from the clean (uncontaminated) demineralized water distribution system (DWDS) by manual isolation valves. However, operating experience has shown that reactor coolant storage and transfer systems can become contaminated with tritium over time, particularly if tritiated water can be recycled, as is the case with the U.S. EPR design (see FSAR Tier 2, Section 9.3.4, “Chemical and Volume Control System [including Boron Recovery System]). This tritiated water can therefore contaminate interfacing clean systems, such as the DWDS, through diffusion, valve leakage, valving errors, or other operating conditions, resulting in the possibility of unmonitored releases of radioactive material to the environment. Therefore, staff requested that the applicant revise the FSAR to include more information on the DWDS, including the location of its storage tanks, any additional contaminated systems it may interface with, the type and location of barriers used to isolate the DWDS from contaminated systems, and the means used to monitor leakage from any connected buried piping, as applicable. **RAI 228, Question 12.03-12.04-10, which is associated with the above requests, is being tracked as an open item**.

In addition to the SSCs described above, there are other structures, systems, and components (SSCs) described in the FSAR which could become contaminated and are, therefore, subject to 10 CFR 20.1406 requirements. Additional details regarding the compliance with 10 CFR 20.1406 can be found in the following sections of this report:

U.S. EPR System	SER Section
New and Spent Fuel Storage	9.1.2
Essential Service Water System	9.2.1
Component Cooling Water System	9.2.2
Ultimate Heat Sink	9.2.5
Condensate Storage Facilities	9.2.6
Seal Water System	9.2.7
Safety Chilled Water System	9.2.8
Raw Water Supply System	9.2.9
Compressed Air System	9.3.1
Process and Post Accident Sampling Systems	9.3.2, 11.5
Ventilation	9.4
Condensate and Feedwater System	10.4.7
Emergency Feedwater System	10.4.9
Radioactive Waste Treatment Systems	11.2 – 11.4

In addition to minimizing contamination of the facility and environment, applicants should incorporate design measures that minimize waste generation and radioactivity during decommissioning. The design has incorporated several lessons learned in this respect. For example, RCS components such as the SGs, RCPs, and the pressurizer can be removed in one piece due to the incorporation of adequately-sized openings. This allows these components to be processed away from the higher dose rates inside containment. In addition, the design of the reactor pit and the spreading compartment located under the reactor allow the reactor pressure vessel to be disassembled underwater, minimizing occupational dose to workers. Any leaks generated during the dismantling will be collected by the IRWST. The design of the U.S. EPR in four physically separated trains allows the dismantling of each train, while keeping in service the auxiliary systems housed in the Fuel Building and the Nuclear Auxiliary Building. The location of the gaseous waste evacuation stack on the roof of the Fuel Building allows the Reactor Building to be disassembled while maintaining the stack in service.

In addition to being a consideration in the design process, meeting the requirements of 10 CFR 20.1406 is an operational concern. Programs and procedures should be in place to minimize contamination of the facility, minimize the contamination of the environment, and facilitate decommissioning. However, the FSAR does not discuss compliance with the operational requirements of 10 CFR 20.1406. **RAI 23, Question 12.03-12.04-1A, which is associated with the above request, is being tracked as an open item**.

12.3.5 Combined License Information Items

The following is a list of item numbers and descriptions from Table 1.8-2 of the FSAR:

Table 12.3-1 U.S. EPR Combined License Information Items

Item No.	Description	FSAR Section	Action Required by COL Applicant	Action Required by COL Holder
12.3-1	A COL applicant that references the U.S. EPR design certification will provide site-specific information on the extent to which the guidance provided by RG 1.21, 1.97, 8.2, 8.8, and ANSI/HPS-N13.1-1999 is employed in sampling recording and reporting airborne releases of radioactivity.	12.3.4.5	Y	
12.3-2	A COL applicant that references the U.S. EPR design certification will provide site-specific information on estimated annual doses to construction workers in a new unit construction area as a result of radiation from onsite radiation sources from the existing operating plant(s). This information will include bases, models, assumptions, and input parameters associated with these annual doses.	12.3.5.1	Y	
12.3-3	A COL applicant that references the U.S. EPR design certification will describe the use of portable instruments, and the associated training and procedures, to accurately determine the airborne iodine concentration within the facility where plant personnel may be present during an accident, in accordance with requirements of 10 CFR 50.34(f)(2)(xxvii) and the criteria in Item III.D.3.3 of NUREG-0737. The procedures for locating suspected high-activity areas will be described.	12.3.4.5	Y	

12.3.6 Conclusions

For the reasons set forth above, the applicant's radiation protection design features help maintain occupational radiation exposures within regulatory limits and ALARA, comply with the requirements of 10 CFR 20.1101(b); the definition of ALARA in 10 CFR 20.1003; the dose limits of 10 CFR 20.1201, 10 CFR 20.1202, 10 CFR 20.1203, and 10 CFR 20.1204; and the non-effluent limits in 10 CFR 20.1301 and 10 CFR 20.1302; and conform to the guidance provided in Regulatory Guides 8.8 and 8.10. In addition, the design features comply with the radiation exposure and radiations 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 design review and from radiation exposure experience gained during the operation of other nuclear power plants. The staff confirmed that these design features are consistent with those contained in Regulatory Guides 8.8 and 8.38 and are, accordingly, acceptable.

The plant design and layout facilitates the control of access to and work within plant areas in accordance with the requirements of 10 CFR 50.34(f)(2)(vii), 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 – Westinghouse Plants (NUREG-1431, Revision 3). Except for the matters identified in **Confirmatory Items 254, Question 12.03-12.04-11, Question 12.03-12.04-12, 12.03-12.04-13, and 12.03-12.04-14**, **RAI 254 , Questions 12.03-12.04-15; RAI 280 , Question 12.03-12.04-17,** and **RAI 296, Question 12.03-12.04-19** the staff finds the plant design and layout acceptable.

The general shield design methodology and source term inventories used by the applicant are similar to those from operating reactors. The basic radiation transport analysis used for the applicant's shield design is based on Microshield 7, MCNP Version 4c, and RANKERN Version 15a. All concrete shielding in the plant will be constructed in general compliance with Regulatory Guide 1.69. Accordingly, except for the matters identified in **RAI 254, Question 12.03-12.04-15** and **RAI 296, Question 12.03-12.04-19**, which request the applicant to provide information to allow the staff to perform confirmatory calculations as well as review the applicant's Quality Assurance program as it was applied to their RANKERN shielding analysis, the staff finds the shielding design and methodology presented in the FSAR acceptable based on the SRP guidelines.

The ventilation system is 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) conforming with all other applicable guidance from RG 8.8. These design criteria, which ensure that that the spread of airborne contamination is minimized or contained, are in accordance with the guidelines of Regulatory Guides 1.52 and 8.8 and, accordingly, demonstrate compliance with the requirements of 10 CFR Part 20.

The applicant's area radiation monitoring system is designed so that it will (1) monitor the radiation levels in areas where radiation levels could become significant and where personnel could be present, (2) alarm when the radiation levels exceed preset levels to warn of increased radiation levels, and (3) display data using the process information and control system. To meet these objectives, the applicant plans to use 41 area monitors located in areas where personnel may be present and where radiation levels could become significant. The design objectives of the 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. The applicant will install airborne radioactivity monitors in the ventilation exhaust ductwork of work areas where there is a potential for airborne radioactivity. These airborne radioactivity monitors will have the capability to detect derived air concentrations (DAC) of the most restrictive particulate and iodine radionuclides in the area or cubicle of lowest ventilation flow rate within 10 hours. Except for the matters identified in **RAI 295, Question 12.03-12.04-18**, the staff finds that the objectives and location criteria of the U.S. EPR area and airborne radiation monitoring systems monitors are in conformance with those portions of 10 CFR 20.1501, 10 CFR 50.34, 10 CFR 50.68, as well as Regulatory Guide 1.97, 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 accident radiation monitors will have emergency power supplies. The systems will be designed for use in the event of an accident in terms of location, usable instrument range and the environment the instrument can withstand, and, accordingly, meet the requirements of 10 CFR 50.34(f)(2)(xvii), Item II.F.1(3) of NUREG-0737, Regulatory Guide 1.97, and BTP 7-10.

On the basis of its review of the information on radiation protection design (including facility design features, shielding, ventilation, and area radiation and airborne radioactivity monitoring instrumentation) supplied by the applicant for the U.S. EPR, as described above, the staff concludes that the applicant has committed to follow the guidelines of the RGs and staff positions set forth in Section 12.03-12.04 of the SRP. Because the FSAR adheres to the guidance provided in these RGs and staff positions, the staff concludes that the relevant requirements of 10 CFR Part 20, 10 CFR Part 50, and 10 CFR Part 70 have been met. The staff finds it acceptable for the applicant to defer discussion of the material addressed by COL Information Items 12.3-1 and 12.3-3. Accordingly, the staff finds the information on radiation protection design contained in FSAR Tier 2, Section 12.3 acceptable except for the matters identified in **Confirmatory Items 254, Question 12.03-12.04-12, 12.03-12.04-13, 12.03-12.04-14; RAI 254 , Question12.03-12.04-15; RAI 280 , Question 12.03-12.04-17, and RAI 296, Question 12.03-12.04-19**

The staff further finds that the dose assessment for the U.S. EPR conforms to the guidelines in RG 1.206 and in Section 12.03-12.04 of the SRP. This dose assessment also conforms to the intent of the guidance in RG 8.19. While the applicant did not conform to all the details in RG 8.19, they provided information on all dose significant activities that occur during normal operations and refueling. By addressing the anticipated occupational radiation exposures due to normal and anticipated inspection and maintenance, and by incorporating design features to reduce occupational radiation exposures, the applicant has shown that the U.S. EPR is designed to operate within the occupational dose limits specified in 10 CFR 20.1201. The staff finds it acceptable for the applicant to defer discussion of the material addressed by COL Information Item 12.3-2. Accordingly, the staff finds the material contained in FSAR Tier 2, Section 12.3.5 acceptable with respect to dose assessment.

The minimization of contamination design philosophy and associated design features conform to the guidelines in RG 4.21, RG 1.206, and in Section 12.03-12.04 of the SRP. The staff has confirmed that the overall design approach, as well as the specific examples of design features (for such systems as the spent fuel pool, effluent discharge line, drain and vent systems, primary and secondary coolant systems, and other auxiliary systems) demonstrate compliance with the guidance of RG 4.21 and the requirements of 10 CFR 20.1406. Accordingly, except for the matters identified in **RAI 23, Question 12.03-12.04-1A, RAI 228 , Question 12.03-12.04-9**, and **RAI 228 , Question 12.03-12.04-10**, the staff concludes that the requirements of 10 CFR Part 20.1406 have been met.

On the basis of the information provided in the U.S. FSAR, Revision 1, on radiation protection design, occupational dose assessment, and minimization of contamination, as described above, the staff concludes that the applicant has committed to follow the guidelines of the RGs and staff positions set forth in Section 12.03-12.04 of the SRP. Because the FSAR adheres to these RGs and staff positions, the staff concludes that the relevant requirements of 10 CFR Part 20, 10 CFR Part 50, and 10 CFR Part 70 have been met, except for the matters identified in **Confirmatory Items 254, Questions 12.03-12.04-11, 12.03-12.04-12, 12.03-12.04-13, and 12.03-12.04-14; RAI 254, Question 12.03-12.04-15; RAI 280 , Question 12.03-12.04-17; RAI 295, Question 12.03-12.04-18, RAI 296, Question 12.03-12.04-19, RAI 23, Question 12.03-12.04-1A; and RAI 228 , Questions 12.03-12.04-9** and **12.03-12.04-10**. The staff finds it acceptable for the applicant to defer discussion of the material addressed by COL Information Items 12.3-1, 12.3-2, and 12.3.-3, since the matters are outside the scope of the requested certification.

12.4 Dose Assessment

The NRC staff's review of this section of the FSAR is documented under Section 12.3 of this report.

12.5 Operational Radiation Protection Program

12.5.1 Introduction

The operational radiation protection program for a nuclear power facility assures that exposures of plant personnel to radiation are controlled and minimized. The administration of the radiation protection program and the qualifications of the personnel responsible for conducting various aspects of the radiation protection program and for handling and monitoring of radioactive material are important components of the program. Adequate equipment, instrumentation, and facilities must also be provided for (1) performing radiation and contamination surveys, (2) in-plant airborne radioactivity monitoring and sampling, (3) area radiation monitoring, and (4) personnel monitoring. Procedures and methods of operation, including those for ensuring that occupational radiation exposure will be as low as is reasonably achievable, must be in place. This includes procedures used in normal operation, refueling, inservice inspections, handling of radioactive material, spent fuel handling, routine maintenance, and sampling and calibration related to radiation safety.

12.5.2 Summary of Application

The applicant states in FSAR Tier 2, Section 12.5 that the subject of this section will be addressed by the combined license applicant.

12.5.3 Regulatory Basis

The relevant requirements of NRC regulations for the operational radiation protection program, and the associated acceptance criteria, are given in Section 12.5, "Operational Radiation

Protection Program," of NUREG-0800. These guidelines in Section 12.5 and the applicable regulatory requirements will be addressed by the NRC staff during the review of combined license applications.

12.5.4 Technical Evaluation

This subject will be addressed by the NRC staff during the review of the combined license applications. However, FSAR Tier 2, Section 12.05, "Operational Radiation Protection Program," COL Information Item 12.05-1, states in part that the COL applicant's description of the radiation protection program will "be consistent" with the applicable regulatory guidance. It is the Staff's position that this language does not convey the applicant's responsibility to describe a program in conformance with the applicable regulatory guides, or to describe and justify acceptable alternatives. Accordingly, the staff has requested that the words "consistent with" be changed to read "conform to" in COL Information Item 12.05-1. **RAI 302, Question 12.05-03, which is associated with the above request, is being tracked as an open item**.

12.5.5 Combined License Information Items

The following is a list of item numbers and descriptions from Table 1.8-2 of the FSAR:

Table 12.5-1 U.S. EPR Combined License Information Items

Item No.	Description	FSAR Section	Action Required By COL Applicant	Action Required By COL Holder
12.5-1	A COL applicant that references the U.S. EPR design certification will fully describe, at the functional level, elements of the Radiation Protection Program. The purpose of the Radiation Protection Program is to maintain occupational and public doses ALARA. The program description will identify how the program is developed, documented, and implemented through plant procedures that address quality requirements commensurate with the scope and extent of licensed activities. This program will comply with the provisions of 10 CFR Parts 19, 20, 50, 52, and 71 and be consistent with the guidance in RG 1.206, RG 1.8, RG 8.2, RG 8.4, RG 8.5, RG 8.6, RG 8.7, RG 8.8, RG 8.9, RG RG 8.10, RG 8.13, RG 8.15, RG 8.27, RG 8.28, RG 8.29, RG 8.34, RG 8.35, RG 8.36, RG 8.38, and the consolidated guidance in NUREG-1736.	12.5	Y	

12.5.6 Conclusions

The staff finds the deferral of the review of this area to the combined license application review and the description of the information item to be acceptable.