12 RADIATION PROTECTION

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12.0 RADIATION PROTECTION

This chapter describes the U.S. Nuclear Regulatory Commission (NRC) staff's review of the radiation protection measures employed by the Advanced Power Reactor 1400 (APR1400), 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 staff evaluated the information in Chapter 12, "Radiation Protection," of the APR1400 design control document (DCD) against the guidance in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (hereafter referred to as the SRP), Chapter 12, "Radiation Protection." Compliance with these criteria provides assurance that doses to workers will be maintained within the occupational dose limits of 10 CFR Part 20, "Standards for Protection Against Radiation." 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 roentgen equivalent man [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 provides the guidance for assuring that radiation doses resulting from exposure to radioactive sources both outside and inside the body can be maintained as required by 10 CFR Part 20, including within the specified limits, 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 and design 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 As Low As (Is) Reasonably Achievable

12.1.1 Introduction

ALARA 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 public health and safety. It also includes using procedures and engineering controls (including adequate design features) based upon sound radiation protection principles.

In addition to providing radiation exposure limits for workers and members of the public, 10 CFR 20.1101, "Radiation protection programs," 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), "Further restrictions on the use of respiratory protection equipment," 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 RG 8.8; RG 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 (DC). The applicant has identified COL information items (see Section 12.1.5 of this safety evaluation report (SER) to ensure that COL applicants referencing the design will address these issues.

12.1.2 Summary of Application

DCD Tier 1: DCD Tier 1, Section 2.8, "Radiation Protection," states that the APR1400 is designed to keep radiation exposures to plant personnel and members of the public within applicable regulatory limits and ALARA.

DCD Tier 2: The applicant has provided a DCD Tier 2, Section 12.1, "Ensuring that Occupational Radiation Exposures Are As Low As (Is) Reasonably Achievable," summarized here in part, as follows:

The majority of nuclear plant worker occupational radiation exposure (ORE) results from maintenance on systems that contain radioactive material, radioactive waste handling, in-service inspection, refueling, abnormal operations, and decommissioning work activities. These activities are addressed and included in the design of the APR1400 through the plant physical layout, selection of materials, shielding, and chemistry control.

During the design process, ALARA design reviews are periodically conducted. The design is based on experience and lessons learned from operating reactors, which indicates that the design of the facility is important to ensuring occupational doses and doses to the public remain ALARA.

Examples of facility design features in the APR1400 design that ensure that the design is ALARA include the separation of radioactive components into individual shielded compartments, remote operating equipment where possible to reduce radiation exposure, and the minimization of field run piping to the extent practicable. A more detailed discussion of design features to ensure exposures to occupational workers and members of the public are ALARA and within applicable dose limits is provided in Section 12.3, "Radiation Protection Design Features" of this safety evaluation report (SER).

The operational aspects of the radiation protection program to provide reasonable assurance that the OREs are ALARA will be provided by the COL applicant, as discussed later in this section.

Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC): There are no ITAAC associated with the review of DCD Tier 2, Section 12.1.

Technical Specifications (TS): There are no TS for this area of review.

12.1.3 Regulatory Basis

The relevant requirements of the Commission's regulations for ensuring that occupational exposures are ALARA, and the associated acceptance criteria, are given in NUREG-0800, Section 12.1, "Assuring that Occupational Radiation Exposures Are As low As Is Reasonably Achievable," and summarized below. No review interfaces with other SRP sections are listed in Section 12.1 of NUREG-0800.

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

1. Section 19.12, "Instruction to workers," of 10 CFR, as it relates to keeping workers who receive ORE informed as to the storage, transfer, or use of radioactive materials or radiation in such areas and instructed as to the risk associated with ORE, precautions and procedures to reduce exposures, and the purpose and function of protective devices employed.

2. Section 20.1101, "Radiation protection programs," of 10 CFR and the definition of ALARA in 10 CFR 20.1003, "Definitions," as they relate to those measures that ensure that radiation exposures resulting from licensed activities are below specified limits and ALARA.

3. Section 52.47, "Contents of applications; technical information," Item (b)(1)of 10 CFR, requires that a design certification (DC) application contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the DC is built and will operate in accordance with the DC, the COL, the provisions of the Atomic Energy Act of 1954, as amended, and the NRC's regulations.

Specific SRP acceptance criteria for the above requirements are as follows:

1. Policy Considerations: Acceptability will be based on evidence that a policy for ensuring that ORE will be ALARA has been formulated in accordance with the training requirements in 10 CFR 19.12," Instruction to workers," and the ALARA provisions of 10 CFR 20.1101(b) and that the policy has been described, displayed, and will be implemented in accordance with the provisions of RG 8.8 (Regulatory Position C.1), RG 8.10 (Regulatory Position C.1) and 10 CFR PART 20, "Standards for Protection Against Radiation," as it relates to maintaining doses ALARA. A specific individual or individuals will be designated and assigned responsibility and authority for implementing ALARA policy. Alternative proposed policies will be evaluated on the basis of a comparison with the above regulatory guides and NUREG-1736, "Consolidated Guidance: 10 CFR Part 20 - Standards for Protection Against Radiation."

2. Design Considerations: Acceptability will be based on evidence that the design methods, approach, and interactions are in accordance with the ALARA provisions of 10 CFR 20.1101(b) and RG 8.8 (Regulatory Position C.2) and will include incorporation of measures for reducing the need for time spent in radiation areas; reducing the frequency of servicing or facilitating maintenance; measures to improve the accessibility to components requiring periodic

maintenance or in-service inspection; measures to reduce the production, distribution, and retention of activated corrosion products throughout the primary system; measures for assuring that ORE during decommissioning will be ALARA; reviews of the design by competent radiation protection personnel; instructions to designers and engineers regarding ALARA design; experience from operating plants and past designs; and continuing facility design reviews. Alternative proposed design policies will be evaluated on the basis of a comparison with the design guidance in RG 8.8 (Regulatory Position C.2).

3. Operational Considerations: Acceptability will be based on evidence that the applicant has a program to develop plans and procedures in accordance with RG 1.33, "Quality Assurance Program Requirements (Operation)," RG , RG 1.8, "Qualification and Training of Personnel for Nuclear Power Plants," RG 8.8, and RG 8.10 that can incorporate the experiences obtained from facility operation into facility and equipment design and operations planning and that will implement specific exposure control techniques.

4. Radiation Protection Considerations: Acceptability will be based on evidence that overall facility operations, as well as the radiation protection program, integrate the procedures necessary to ensure that radiation doses are ALARA, including work scheduling, work planning, design modifications, and radiological considerations.

12.1.4 Technical Evaluation

The staff reviewed the information in DCD Tier 2, Section 12.1, to assess adherence to the guidelines and criteria in SRP Section 12.1, regarding the radiation protection aspects of the reactor design. Specifically, the staff reviewed DCD 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 SRP Section 12.1, or had provided acceptable alternatives. As described below, the staff determined that DCD, Tier 2, Section 12.1, conforms to the applicable guidance contained in these RGs and applicable staff positions with the exception of the confirmatory items discussed below.

12.1.4.1 Policy Considerations

In DCD 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 state of the APR1400 design process. The applicant has committed to ensure that the APR1400 plant will be designed and constructed in a manner consistent with the guidelines of RG 8.8. In particular, DCD 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 lessons learned from the nuclear power industry, including operating experience from pressurized water reactors in the United States and Korea.

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, for both occupational workers and members of the public. The detailed policy considerations regarding overall plant operations and implementation of such a radiation protection program are outside the scope of the DC review. In order to maintain doses to plant personnel ALARA, DCD Tier 2, Section 12.1.3, “Operational Considerations,” states that the COL applicant will submit a description of the radiation protection program to provide reasonable assurance that ORE will be ALARA. However, none of the COL items in DCD Section 12.1 or Section 12.5, “Operational Radiation Protection Program,” specify that the program will ensure that doses to members of the public will be ALARA, therefore, the staff issued Request for Additional Information **(RAI) 213-8222, Question 12.01-1 (ML15258A039)**, requesting that the applicant clarify how the COL items addressing the radiation program will also ensure the public dose will be ALARA.

In its response to **RAI 213-8222, Question 12.01-1, (ML15336A999)**, the applicant proposed updating DCD COL Information Item 12.1(2) to specify the radiation protection program will not only assure that OREs are ALARA, but will also assure that public radiation exposures are ALARA. Since the COL action items address occupational and public radiation exposure in accordance with 10 CFR 20.1101(b), RG 8.8 and 8.10, and NUREG-1736, the staff finds this acceptable pending the proposed DCD updates. Therefore, **RAI 213-8222, Question 12.01-1, is being tracked as a confirmatory item**.

In order to maintain doses to plant personnel ALARA, in COL Information Item 12.1(1) and COL Information Item 12.1(3), the application also states that the COL applicant will describe how the plant follows the guidance of the following RGs, which comply with the requirements of 10 CFR Part 20:

- RG 1.33, Revision 2, “Quality Assurance Program Requirements (Operation).”
- RG 1.8, Revision 3, “Qualification and Training of Personnel for Nuclear Power Plants.”
- RG 8.2, Revision 1, “Guide for Administrative Practices in Radiation Monitoring.”
- RG 8.4, “Personnel Monitoring Device - Direct-Reading Pocket Dosimeters.”
- RG 8.7, Revision 2, “Instructions for Recording and Reporting Occupational Radiation Exposure Data.”
- RG 8.8, Revision 3, “Information Relevant to Assuring that Occupational Radiation Exposures at Nuclear Power Stations Will be as Low as Is Reasonably Achievable.”
- RG 8.9, Revision 1, “Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program.”
- RG 8.10, Revision 2, “Operational Philosophy for Maintaining Occupational Radiation Exposures as Low as Is Reasonably Achievable.”
- RG 8.13, Revision 3, “Instruction Concerning Prenatal Radiation Exposure.”
- RG 8.15, Revision 1, “Acceptable Programs for Respiratory Protection.”
- RG 8.20, Revision 2, “Applications of Bioassay for Radioiodine.”
- RG 8.25, Revision 1, "Air Sampling in the Workplace.”
- RG 8.26, “Applications of Bioassay for Fission and Activation Products.”
- 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, Revision 1, “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.”

In addition to the above, SRP Section 12.5 specifies that the staff will review information describing the implementation of RG 8.4, “Personnel Monitoring Device – Direct-Reading Pocket Dosimeters,” therefore, in **RAI 6-7854, Question 12.05-1 (ML15155B329)**, the staff requested that the applicant clarify how the guidance of RG 8.4 will be met or provide an alternative approach. In its response to **RAI 6-7854, Question 12.05-1, (ML15166A246)**, the applicant indicated that COL Information Item 12.1(3) will be updated in a future DCD revision to indicate that the COL applicant will also describe how the plant follows the guidance provided in NRC RG 8.4. The staff determined that this approach is acceptable. Therefore, **RAI 6-7854, Question 12.05-1 is being tracked as a confirmatory item**.

Compliance with 10 CFR Part 19, “Notices, Instructions and Reports to Workers: Inspection and Investigations,” requires that workers who receive occupational exposure be kept informed of radioactive material and radiation and receive instructions with the objective of minimizing exposures to radioactive materials. As indicated, COL Information Item 12.1(3) specifies that the COL applicant will specify how the plant follows RG 8.27. RG 8.27 addresses the requirements of 10 CFR 19.12.

12.1.4.2 Design Considerations

The plant radiation protection design should ensure that individual doses and total person-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. DCD Tier 2, Section 12.1.2, “Design Considerations,” describes the objectives for the general design. These include design features to satisfy ALARA objectives and include the following:

- Highly radioactive equipment is separated from less radioactive equipment to the extent practicable.

- Major components containing high levels of radioactivity are provided with their own cubicles and adequate shielding to minimize exposure during maintenance.

- Adequate spacing is provided to facilitate maintenance of components and to allow for the installation of temporary shielding, if needed.

- Labyrinths are provided for entrances to highly radioactive areas to limit the effects of direct or scatter radiation to the areas outside the entrance way.

- Sufficient space is provided for ingress and egress, equipment laydown/pull areas, and removal paths for major equipment that is expected to be replaced during the plant lifetime.

- Pumps and valve galleries are provided outside the cubicles containing the radioactive equipment to minimize operational exposures during operation and maintenance activities.

- Piping that contains contaminated fluid is routed inside pipe chases to the maximum extent practicable.

- Shield wall penetrations are positioned to minimize radiation streaming from radioactive equipment to low-radiation areas.

- Instruments are located in low-radiation areas, and remote operation is implemented to the maximum extent practicable.

- Components are designed with minimal cobalt content to the extent practicable, to reduce the amount of activation products.

- Components with the potential for crud accumulation are designed to prevent the settlement of crud through the use of ellipsoidal bottoms. These systems are also provided with a flushing capability to remove any crud that does deposit.

- The ventilation systems are designed so that the air flows from clean areas to potentially contamination areas to minimize the potential for the spread of airborne contamination.

The design features described in DCD 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 determined that the general design features are acceptable.

In accordance with Chapter 12 of the SRP, in Section 12.1, the staff reviewed the design methods and approach to ensure that they are in accordance with 10 CFR 20.1101(b) and RG 8.8. More detailed and specific radiation protection design features are provided in DCD Section 12.3 and will be discussed and evaluated in more detail in Section 12.3, "Radiation Protection Design Features (including Dose Assessment)" of this SER, in accordance with SRP Section 12.3-12.4.

12.1.4.3 Operational Considerations

Operational considerations regarding the implementation of a radiation protection program are outside the scope of this DC review. The applicant has stated that a COL applicant who references the APR1400 certified design will address operational and maintenance requirements, while satisfying the guidance of RG 1.33, RG 1.8, RG 8.8, and RG 8.10. The applicant also listed other RGs that the COL applicant will need to address in its application, which are discussed under the policy considerations section above. It is acceptable for COL applicants to address the operational and maintenance considerations as described in the COL items applicable to this section.

12.1.4.4 Radiation Protection Considerations

The ALARA program is part of the radiation protection program. The radiation protection program is evaluated in Section 12.5 of this SER.

12.1.5 COL Information Items

The following is a list of items from DCD, Tier 2, Table 1.8-2, "Combined License Information Items":

- COL Information Item 12.1(1): The COL applicant is to provide the organizational structure to effectively implement the radiation protection policy, training, and reviews consistent with operational and maintenance requirements, while satisfying the applicable regulations and RGs including NRC RG 1.33, RG 1.8, RG 8.8, and RG 8.10.

- COL Information Item 12.1(2): The COL applicant is to describe the operational radiation protection program to provide reasonable assurance that OREs are ALARA.

- COL Information Item 12.1(3): The COL applicant is to describe how the plant follows the guidance provided in NRC RGs 8.2, 8.7, 8.9, 8.13, 8.15, 8.20, 8.25, 8.26, 8.27, 8.28, 8.29, 8.34, 8.35, 8.36, and 8.38.

12.1.6 Conclusion

Based on the information supplied by the applicant, as described above, the staff concludes that with the incorporation of the above confirmatory items into the DCD, the general APR1400 design features meet the criteria of Section 12.1 of the SRP. These design features are intended to maintain individual doses and total person-rem doses within the limits of 10 CFR Part 20 and ALARA. Therefore, the staff determined that the general APR1400 design features and commitments are acceptable. The COL applicant will address the policy and operational considerations for the APR1400. The staff determined that it is acceptable for the applicant to defer discussion of the material addressed by the COL information items to the COL applicant. The staff will determine compliance with the requirements of 10 CFR Part 20 in these areas during the COL application review.

12.2 Radiation Sources

12.2.1 Introduction

The projected radiation sources during normal operations, anticipated operational occurrences (AOOs), and accident conditions in the plant, are used as the basis for radiation shielding design and other radiation protection design features and procedural requirements. This includes the location of sources in the plant, the isotopic composition, 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. It also provides in-plant radiation sources that could be present during design basis accident (DBA) conditions.

12.2.2 Summary of Application

DCD Tier 1: In its response to **RAI 23-7929, Question 12.02-6 (ML15223B087)** the applicant proposed updating Tier 1, Table 2.7.4.4-1, "Light Load Handling System Equipment Location

Characteristics," to specify where fuel will be handled and stored in the APR1400 design. This information is pertinent to the staff's review of the shielding and controls for radiation sources.

In its response to **RAI 308-8339, Question 12.02-19 (ML16272A470)**, the applicant proposed updating DCD Tier 1, Section 2.4.6.1, "Design Description," to provide information on the decontamination factor for the pre-holdup ion exchanger, which is used to reduce the source terms for downstream components. In its response to **RAI 343-8420, Question 12.02-22 (ML16279A524)**, the applicant proposed adding information to Tier 1, Section 2.4.6.1 to describe how the chemical volume and control system (CVCS) gas stripper is used to control radioactive source terms for the holdup tank and effluent releases.

The information provided in these RAI responses is evaluated in Section 12.2.4 of this SER.

DCD Tier 2: The applicant has provided a Tier 2 system description in Section 12.2, "Radiation Sources," of the DCD, which is summarized here, in part, as follows:

Section 12.2 discusses and identifies the sources of radiation that form the basis for the shielding design calculations, radiation zoning, and to perform dose assessments. In addition, it describes the sources of airborne radioactivity used to design personnel protection measures. Finally, it provides in plant radiation sources that could be present during design-basis accident (DBA) conditions, including shutdown cooling, containment spray, and safety injection system source terms, during accident conditions. These systems are used to recirculate post-accident fluid for core cooling during certain DBAs, including a loss of coolant accident (LOCA). It also provides the source term for the main control room (MCR) emergency filters during accident conditions. These accident radiation sources are used to determine the dose to the operators in the MCR, the dose to operators in accessing vital areas that operators may be required to access to shut down the reactor or maintain safe shutdown during an accident, and to determine the accident doses to equipment included in the environmental qualification (EQ) program, as discussed in Section 3.11, "Environmental Qualification of Mechanical and Electrical Equipment," of this SER.

The shielding design-basis for primary coolant source term is based on APR1400-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.

The sources of radiation during normal full-power operation are direct core radiation, coolant activation, the activation of the reactor coolant corrosion and erosion products, and the leakage of fission products from defects in the fuel rod cladding. This radioactive material is continuously transported through the reactor coolant piping. During operation, nitrogen-16 (N-16), which is formed by neutron interaction with oxygen-16 in water, is the largest source of radioactivity in the reactor coolant system (RCS) piping, reactor coolant pumps, and steam generators (SGs), and, because of its 6 mega-electron-volt (MeV) decay gamma radiation, N-16 has the most impact on the shielding design for the RCS during normal plant operation. New N-16 is continuously produced as coolant passes through the reactor core. The amount of N-16 activity in each of the primary coolant system components depends on the total transit time to the component and the residence time in the component. N-16 activity is generally not a concern for radiation source term for systems and components located outside the containment due to its short (7.13 seconds) half-life and the transport time from the reactor vessel to outside of containment. However, it was initially unclear whether the CVCS piping before leaving the containment provides an adequate decay time to neglect the contribution from N-16 outside of the containment and that the design included proper shielding for N-16 source term. Therefore, the staff requested additional information on this topic, as discussed in Section 12.3 of this SER.

Fission and corrosion product activities circulating in the reactor coolant system (RCS) and other systems and components comprise the remaining significant radiation sources during full-power operation. During plant operation, radioactive non-gaseous fission and corrosion products deposit on the inner surfaces 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). The design basis of 0.25 percent failed fuel fraction is used as the source term to establish shielding provisions for systems and components that may contain radioactive material. The most significant systems and components are located inside containment, the auxiliary building and the compound building. The compound building houses radwaste processing system components. The fission and corrosion product activities circulating in the reactor coolant from which these source terms are based are given in APR1400 DCD Tier 2, Table 12.2-5, "Reactor Coolant Equilibrium Concentration (Core Power: 4,063 Mega-Watts-thermal (MWt), 0.25% Fuel Defect, No Gas Stripping)." The reactor coolant source term is calculated using the DAMSAM code, ("DAMSAM: A Digital Computer Program to Calculate Primary and Secondary Activity Transients," Combustion Engineering, Inc.," 1972). Values of corrosion product concentrations are based, in part, on operating pressurized water reactor (PWR) reactor data, as provided in American National Standards Institute/American Nuclear Society (ANSI/ANS) Standard 18.1-1999, "Radioactive Source Term for Normal Operation of Light Water Reactors." Dimensions and parameters for contained radiation sources, including source composition and source housing material and thicknesses, are provided in DCD Tier 2, Table 12.2-25, "Radioactive Source Dimensions and Parameters Used in Shielding Analysis."

Airborne radioactivity concentrations can occur in the containment building, both during power operation (coolant leakage) and refueling (evaporation of the refueling pool). The spent fuel pool (SFP) water contains radionuclides from defects in spent fuel and corrosion products from fuel assemblies. The evaporation of the SFP water can lead to airborne radioactivity concentrations in the SFP area of the auxiliary building both during power operation and refueling. The fuel handling area HVAC system serves the spent fuel pool area of the auxiliary building separately from the auxiliary building HVAC systems.

Airborne radioactivity concentrations within the reactor building during operation, areas of the auxiliary building (other than the SFP area), and in the compound building result principally from equipment leakage. The design of the ventilation systems in radiological portions of these buildings provides for airflow that is from regions that are expected to have a lower potential for airborne contaminates (this includes hallways and other areas expected to be routinely accessed, as well as other areas with minimal radioactive sources present) to those with higher potential for airborne contaminates (such as hot pipe chases and areas with high activity radiation sources that are susceptible to leakage). Areas with a higher potential for airborne contaminates are not routinely occupied. Since the airflow is generally from areas with a low potential for airborne radioactivity (such as those areas that are frequently accessed) to areas with a higher potential for airborne radioactivity, the airborne activity concentrations are expected to be low in those areas of the buildings that are routinely occupied and airborne radioactivity in highly contaminated areas is not expected to spread to clean areas.

Under normal operating conditions (no leakage into the component cooling water [CCW] system (CCWS)), components within the uncontrolled portions of the auxiliary building are not expected to contain radioactive material. Airborne radioactivity concentrations in the uncontrolled portions of the auxiliary building are, therefore, expected to be negligible.

Under normal operating conditions (no primary-to-secondary leaks and no leakage into the CCWS), components within the turbine building are not expected to contain high levels of radioactive material. As a result, airborne radioactivity concentrations in the turbine building are expected to be negligible. In addition, the design includes a separate structure near the essential service water piping to house CCW heat exchangers. During normal operation, this structure is also not anticipated to contain high levels of radioactive material.

ITAAC: There are no ITAAC associated with the review of DCD Tier 2, Section 12.2.

TS: There are no TS for this area of review.

12.2.3 Regulatory Basis

The relevant requirements of the NRC's regulations for this area of review, and the associated acceptance criteria, are in Section 12.2, "Radiation Sources," of NUREG-0800, and are summarized below. No review interfaces with other SRP sections are listed in Section 12.2 of NUREG-0800.

1. Section 20.1101, "Radiation protection programs", of 10 CFR, 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. Section 20.1203, "Determination of external dose from airborne radioactive material," of 10 CFR 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. Section 20.1301, "Dose limits for the individual member of the public," of 10 CFR, as it relates to the determination of radiation levels and radioactive materials concentrations and public exposure limits.

4. Part 50, "Domestic Licensing of Production and Utilization Facilities," of 10 CFR, Appendix A, "General Design Criteria for Nuclear Power Plants," General Design Criteria 61, " Fuel storage and handling and radioactivity control," as it relates to systems that may contain radioactive materials.

5. Section 50.34(f)(2)(vii) of 10 CFR 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 those conditions.

6. Section 52.47(a)(5) of 10 CFR, which requires that the DC application must include the kinds and quantities of radioactive materials expected to be produced in the operation and the means for controlling and limiting radioactive effluents and radiation exposures within the limits set forth in 10 CFR Part 20.

7. 10 CFR 52.47(a)(22) as it relates to ensuring that information necessary to demonstrate how operating experience insights have been incorporated into the plant design.

8. Section 52.47(b)(1) of 10 CFR, which requires that a DC application must include the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the DC is built and will operate in accordance with the DC, the COL, the provisions of the Atomic Energy Act of 1954, as amended, and the NRC's regulations.

The following RGs, standards, and NUREGs provide information, recommendations, and guidance and in general, describe a basis acceptable to the staff for implementing the requirements of 10 CFR 52.47(a)(5), 10 CFR 20.1201, 10 CFR 20.1202, 10 CFR 20.1203, 10 CFR 20.1204, 10 CFR 20.1206, 10 CFR 20.1207, "Occupational dose limits for minors," 10 CFR 20.1301, and 10 CFR 20.1801, "Security of stored material."

1. RG 1.183, Revision 0, "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 LOCA and for determining the design basis radionuclide concentrations during other accidents, that may be more limiting than the LOCA in certain plant areas.

2. RG 1.7, Revision 3, "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, Revision 1, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors," as it relates to complying with 10 CFR 20.1301 concerning 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 18.1-1999, "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 DCD Tier 2, Section 12.2, "Radiation Sources," to assess conformance with the guidance in RG 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)," and the criteria in Section 12.2 of the SRP. The applicant used the contained source terms described in the DCD as the basis for the radiation shielding design calculations and personnel dose assessment. The applicant used the airborne radioactive source terms in the DCD in the design of ventilation systems and for assessing personnel dose. The staff reviewed the source terms in the DCD to ensure that the applicant had either committed to follow the guidance of the RGs and staff positions set forth in Section 12.2 of the SRP, or provided acceptable alternatives, which are further described in the evaluations of the specific issues.

12.2.4.1 Contained Sources

In DCD Tier 2, Section 12.2.1, "Contained Sources," the applicant described the source terms used for determining the radiation shielding and zoning during normal full-power operation, including AOOs, such as refueling. DCD Tier 2, Section 12.2.1 describes all large contained sources of radiation that are used as the basis for designing the radiation protection program and completing shield design calculations. These sources include the reactor core; the RCS; the CVCS; the liquid, gaseous, and solid radioactive waste systems; and other miscellaneous sources. For each of these contained sources, the applicant provided either the source strength by energy group or the associated maximum activity levels listed by isotope (in most cases the source terms are provided by activity level).

The sources of radiation during normal full-power operation are direct core radiation, coolant activation processes, the leakage of fission products from defects in the fuel rod cladding, and the activation of the reactor coolant corrosion and erosion products.

The DAMSAM code is used to calculate the design basis fission product inventory in the fuel pellet region and the reactor coolant region, as discussed in DCD Subsection 11.1.1.1, "Fission Product Activities in the Reactor Coolant." While the staff has not reviewed and approved the DAMSAM code, the information provided by the code is acceptable for the APR1400 design because the code methodology appears comparable to the methodology used in other design applications. In addition, the core source term and reactor coolant source terms provided for the APR1400 design are comparable to other new PWR reactor designs when accounting for differences in power level. Finally, using the maximum core source term values in DCD Table 15A-1 and the information provided in DCD Table 12.2-5, staff estimated similar RCS radionuclide concentrations for several radionuclides, as those provided in DCD Table 12.2-5.

Sources downstream of the RCS were calculated using the Shield-APR code (A proprietary computer code developed by KHNP to facilitate the estimation of shielding and source inventories for the CVCS and its interface systems/components within the scope of APR1400 design.) and DIJESTER code, ("A Program to Compute Radioactive Decay in Fluid Flow Systems," D. J. Pichurski, April 1976). The DIJESTER code was used for liquid waste management system (LWMS) sources, and the Shield-APR code was used for other sources. The Shield-APR code does not account for the buildup of daughter progeny within components. This issue is discussed later (see the discussions related to RAI 142-8090, Question 12.02-13 and RAI 343-8420, Question 12.02-23, below). In addition to the contained sources provided in the DCD and discussed below, DCD Section 12.2.4, "Combined License Information," contains COL Information Item 12.2(1), which indicates that, "The COL applicant is to provide any additional contained radiation sources, such as instrument calibration radiation sources, that are not identified in Subsection 12.2.1." The staff agrees that it is acceptable and appropriate for the COL applicant to provide information on any site-specific contained sources.

12.2.4.1.1 The Reactor Core and Fuel

Neutron and gamma sources emanating from the reactor core during normal operation and shutdown are provided in DCD Tables 12.2-1, “Normal Operation Neutron Spectra Outside the Reactor Vessel,” 12.2-2, “Normal Operation Gamma Spectra Outside the Reactor Vessel, and 12.2-3, “Shutdown Gamma Spectra Outside the Reactor Vessel.” This information is based on core parameters provided in DCD Tier 2, Chapter 4, “Reactor,” and are, therefore, acceptable. DCD Tier 2, Table 12.2-8, “Parameters Used in Spent Fuel Decay Gamma Source Calculation,” provides parameters used in determining the source term for the core after a cycle of operation, which can be used to determine dose rates during fuel movement and in the SFP. DCD Tier 2, Table 12.2-9, “Spent Fuel Gamma Source,” provides the core source term at various times post shutdown. This information can be used to determine dose rates during fuel movement and in the SFP by converting the source term of the entire core to an individual assembly (considering the peaking factor of the maximum assembly).

However, the staff identified the following problems with the information provided in the DCD regarding the spent fuel source term:

1. The DCD discussed only the gamma source term in the spent fuel and did not include a neutron dose contribution nor discuss the dose contribution from neutron radiation.

2. The DCD did not adequately describe how the source term in Table 12.2-9 was used to calculate shielding around the SFP.

3. The DCD indicated that shielding for the spent fuel transfer tube was based on fuel 100 hours after operation but did not provide a basis for the 100-hour value.

4. The DCD did not adequately describe how the dose to an operator on the refueling machine or spent fuel handling machine was determined.

5. There was an apparent inconsistency between the text and DCD figures regarding the dose to an operator on the refueling platform.

Consequently, the staff issued **RAI 7-7855, Question 12.02-1 (ML15134A482)**, to request clarification from the applicant of the above issues. In its response to **RAI 7-7855, Question 12.02-1 (ML15166A287)**:

> The applicant provided neutron source term information. However, since the neutron dose contribution was negligible compared to gamma radiation, it was not included in the DCD. Because the neutron radiation is negligible for shielding and zoning purposes, it is acceptable to not include the neutron source in the DCD.
>
> The applicant indicated that the SFP shielding is based on the maximum capacity of the SFP assuming all fuel was recently removed from the core. This approach is conservative and is, therefore, acceptable.
>
> The applicant indicated that all shielding and dose calculations for the SFP and for spent fuel transfer were based on moving fuel 100 hours after shutdown. The 100-hour timeframe was based on operating data from Korean reactors. However, the staff's

review of TS 3.9.3, “Containment Penetrations,” and TS 3.9.6, “Refueling Pool Water Level” indicates that these specifications are based on core alterations being performed 72 hours after shutdown. Since TS allow fuel to be moved 72 hours after shutdown, the staff cannot accept the applicant’s approach of basing the design basis shielding design for fuel transfer on an assembly being moved 100 hours after shutdown. This is based on the staff's analysis that shows the spent fuel source term could be significantly higher (possibly more than 30 percent higher), if the spent fuel is being moved at 72 hours instead of 100 hours.

The applicant indicated that the dose to an operator on the refueling machine and spent fuel handling machine are based on fuel raised to the maximum height allowed and that a limit switch will prevent lifting the assembly above that height and that the dose to an operator would remain below 0.0025 rem (Roentgen equivalent man) per hour, or 2.5 mrem/hr (25 microsieverts [μSv]/hr. The applicant proposed DCD updates to clarify this information. Because the information provided is consistent with ANSI/ANS 57.1-1992, "Design Requirements for Light Water Reactor Fuel Handling Systems," which the applicant references and is also referenced in the SRP, the staff finds that the applicant's information and the proposed DCD markups are acceptable.

The applicant proposed to update the DCD Chapter 12 figures to provide a footnote indicating that the dose on the refueling machine platform was a maximum of 2.5 mrem/hour (25 μSv/hr), consistent with ANSI/ANS 57.1-1992, which is acceptable.

Since the source terms for calculating shielding, dose rates, and zoning for spent fuel were based on 100 hours instead of 72 hours, as is provided in the TS (as discussed above), the staff closed **RAI 7-7855, Question 12.02-1 as an unresolved item** and issued **RAI 146-8152, Question 12.02-14 (ML15222B315)**. In RAI 146-8152, Question 12.02-14, the staff requested that the applicant address this issue of not basing the shielding and zoning on the maximum allowed source term and to resolve the discrepancy between the two post-operation time periods. In its response to **RAI 146-8152, Question 12.02-14, (ML15344A155)**, the applicant proposed revising the surveillance requirements and basis for TS 3.9.3.1 and TS 3.9.6, to indicate that the decay time before fuel movement was 100 hours. In addition, in its response to **RAI 133-7978, Question 16-31, (ML16036A378)**, Part 15, the applicant proposed adding TS 3.9.8, “Decay Time,” which specifies that fuel cannot be moved until the reactor has been subcritical for 100 hours. Based on the new TS 3.9.8 that requires fuel movement not to begin until 100 hours after the reactor is subcritical, the staff finds it acceptable to base the shielding design for the refueling pool, fuel transfer tube, refueling canal, and the SFP, as well as the shielding analysis for a raised fuel assembly in the refueling pool and SFP area on fuel being moved 100 hours after being subcritical. Therefore, **RAI 146-8152, Question 12.02-14 is being tracked as a confirmatory item**, pending the DCD changes specified in the response to RAI 146-8152, Question 12.02-14, as well as the addition of TS 3.9.8, as specified in the response to RAI 133-7978, Question 16-31 (ML16036A378). This is being tracked as a **confirmatory item**. A review of the shielding design and dose to operators during spent fuel transfer is provided in Section 12.3 of this SER.

Also, it was unclear to the staff if the control element assembly change platform and elevator or other areas or equipment besides the equipment specifically designated for handling fuel could be used to transport or store spent fuel. In addition, information describing fuel storage areas is also pertinent in criticality analysis, potential drain down events, and in the seismic design. Therefore, the staff issued **RAI 23-7929, Question 12.02-6 (ML15174A324)**, requesting that the applicant describe all areas where fuel will be handled or stored and update Tier 1 to clarify where fuel will be handled and stored. In its **response to RAI 23-7929, Question 12.02-6, (ML15223B087)**, the applicant updated Tier 1, Table 2.7.4.4-1 to specify that the only equipment or locations used to handle or store fuel (other than the new fuel racks, spent fuel racks, reactor core, and the fuel transfer system used to transport fuel assemblies between the SFP and containment) are the refueling machine, spent fuel handling machine, new fuel elevator, and fuel handling hoist of overhead crane, and that no other equipment, locations, or areas will be used to store or handle fuel at APR1400 plants. Based on the proposed change to Tier 1 that clarifies which equipment and locations are used to handle or store fuel and the areas identified were those which staff expected, there are no new areas needing to be evaluated for fuel storage or handling for radiation protection purposes and the response is acceptable within the scope of this SER section. **RAI 23-7929, Question 12.02-6 is being tracked as a confirmatory item** pending the proposed DCD revisions**.** The detailed review of fuel handling areas and locations can be found in other sections of the staff's SER, including Chapter 9, "Auxiliary Systems." In addition, the plant radiation shielding and zoning, as it relates to fuel transfer is discussed in Section 12.3.

DCD Tier 2, Table 12.2-25, provides cask loading pit source dimensions, which appear to be the dimensions of a single fuel assembly. It was unclear if the applicant intended to store fuel in the cask loading pit area, other than during cask loading. In addition, it was unclear which source term was used to determine the cask loading pit shielding thicknesses provided in DCD Tier 2, Table 12.3-4, "Design Basis Radiation Shield Thicknesses," (for the walls and floor to the cask loading pit area). The staff issued **RAI 23-7929, Question 12.02-8 (ML15174A324)**, requesting that the applicant provide this information. In its response to **RAI 23-7929, Question 12.02-8, (ML15223B087)**, the applicant proposed adding information to the DCD specifying that for the cask loading pit shielding analysis, it was assumed that a single spent fuel assembly is being loaded into the cask within the cask loading pit. The single fuel assembly is conservatively assumed to have decayed for 1,000 hours after reactor shutdown. Since the spent fuel is typically at least five years old before cask transfer and the NRC has never authorized fuel to be transferred to a cask in less than three years after being removed from the core, the 1,000-hour assumption is conservative. In addition, the cask itself would provide shielding for other fuel assemblies placed in the cask during cask loading. Therefore, there will never be more than one assembly in the cask transfer area that is not within the cask. Casks are designed to provide suitable shielding for personnel in the area during cask transfer, including when the cask is loaded and outside of the fuel pool (with appropriate procedural controls to keep exposures ALARA). Therefore, the dose rate through the cask loading pit shield walls and floor from the fuel within the cask would be negligible compared to a 1,000-hour old fuel assembly being transferred. As a result, the applicant's assumptions for the cask loading pit shielding analysis and shielding thicknesses for the cask transfer area are acceptable. Therefore, **RAI 23-7929, Question 12.02-8 is being tracked as a confirmatory item**, pending the proposed DCD additions.

12.2.4.1.2 Reactor Coolant System and Contained Sources Other Than Fuel

All sources, other than reactor fuel and activated components, are based on an assumed 0.25 percent failed fuel. As discussed in SER Section 12.2.4.1 above, the applicant calculated the RCS source term, assuming a 0.25 percent fuel defect, using the DAMSAM code. However, for several radionuclides, mostly corrosion products, the applicant reported that the RCS source term was lower than the expected values based on ANSI/ANS-18.1-1999. The applicant increased the source term for these radionuclides to values based on ANSI/ANS-18.1-1999. Because this adjustment, to base the source term on ANSI/ANS-18.1-1999, is consistent with the SRP, this general approach is acceptable. The 0.25 percent failed fuel source term is provided in DCD Table 12.2-5. However, a review of the application reveals that DCD Table 15A-1, "Maximum Core Fission Product Inventories (Core Power: 4,062.66 MWt, Burnup : 56.4 Giga-Watt-Days per Metric Ton of Uranium (GWD/MTU))," provides fission product inventories of 60 essential radionuclides, yet the RCS source term and source terms for most contained sources did not include some of these radionuclides. In addition, some sources, including waste management system sources included radionuclides that were not provided in either DCD Table 15A-1 or Table 12.2-5. Therefore, the staff issued **RAI 103-7998, Question 12.02-10 (ML15205A401),** requesting that the applicant provide additional information about radionuclides missing from the RCS source term and source terms for most contained sources. In its response to **RAI 103-7998, Question 12.02-10, (ML15245A493)**, the applicant clarified that all source terms include the radionuclides listed in ANSI/ANS-18.1-1999 and that the liquid waste management system (LWMS) and solid waste management system (SWMS) include additional radionuclides because the DIJESTER code, which was used to calculate the source terms for the LWMS included contributions for additional radionuclides. SRP Section 12.2 indicates that "the quantities will be acceptable if the specific values given in the tables are consistent with ANSI/ANS-18.1 and RG 1.112 for coolant and corrosion activation product source terms." Therefore, the radionuclides that the applicant included are acceptable. **RAI 103-7998, Question 12.02-10, is resolved and closed**.

In reviewing the source term information in the application and information provided during the source term audit (see ML15208A492 for the audit plan and ML16119A083 for the audit report), the staff determined that the applicant was not properly accounting for the buildup of radionuclide daughters in components downstream of the RCS. Specifically, a review of the Shield-APR code manual, which is used to calculate the source terms for most of the components downstream of the RCS, indicated that the Shield APR code does not account for the buildup of daughter radionuclides from the decay of the parent within source media. Not accounting for the buildup of daughter radionuclides from the decay of the parent radionuclides can significantly affect the radionuclide inventory within components for radionuclide daughters with shorter half-lives because the inventory of those radionuclides within components is based largely on the decay of the parents.

While the Shield APR code does not account for the buildup of daughter radionuclides, the applicant modified the source term to account for barium-137m (Ba-137m) generation from the decay cesium-137 (Cs-137) (by making the quantity of Ba-137m within sources the same as the value of Cs-137, which is an acceptable assumption. However, for other nuclides, the source terms provided do not appear to account for daughter nuclide buildup at all. For example, in the spent resin long-term storage tank, which accumulates CVCS resin over a period of years, the applicant reports values of zero Becquerel's (Bq) for rhodium-106 (Rh-106), praseodymium-143 (Pr-143), and yttrium-90 (Y-90), even though there are large quantities of their parents decaying for years within the tank.

In order to resolve this issue, the staff **issued RAI 142-8090, Question 12.02-13 (ML15221A004),** requesting that the applicant either provide justification for not including the radionuclide daughters or revise their methodology for calculating source terms to include consideration for the buildup of daughter products and to revise DCD Chapters 11 and 12 to provide updated information accounting for daughter buildup.

In its response to **RAI 142-8090, Question 12.02-13, (ML15286A242)**, the applicant indicated that downstream of the RCS, the only sources to consider the buildup of daughter products are the LWMS and solid waste management system (SWMS) sources (except for Ba-137m, which is considered to be in equilibrium with Cs-137 for most sources). The applicant indicated that the RCS design bases source terms for each nuclide activity were determined by choosing the maximum value during a five cycle operating period and that, since all shielding sources are based on this, all the RCS design bases source terms are conservative. However, the applicant provided no explanation or data supporting the claim that the source terms were more conservative than if the buildup of daughters were considered. In addition, the staff's experience has been that the 0.25 failed fuel percent RCS source terms normally reach near maximum values after just several months of operation and the total source strength would change very little after that time. Therefore, the **staff closed RAI 142-8090, Question 12.02-13, as an unresolved item** and issued **RAI 343-8420, Question 12.02-23 (ML15356A016)**, requesting that the applicant revise source terms, shielding, and zoning, as appropriate, to consider the buildup of radioactive daughters, for those radionuclides listed in ANSI/ANS 18.1 (consistent with SRP Section 12.2) or provide additional justification for why the current source terms are acceptable.

In its response to **RAI 343-8420, Question 12.02-23, (ML16274A461)**, the applicant indicated that, while most radionuclides reach equilibrium in one cycle, for the APR1400 design, several radionuclides do not reach equilibrium until the fourth or fifth cycle, which justifies why the applicant used five cycles to calculate the maximum design basis RCS source terms. This response, however, provides no justification for not considering daughter progeny downstream of the RCS.

In its response to **Question 12.02-23 (ML16274A461)**, the applicant also indicated that Westinghouse would be comparing the DAMSAM and Shield-APR codes to Westinghouse codes, which are used to perform similar calculations. The Westinghouse codes consider the buildup of daughter progeny and have been used to develop source terms for the currently approved AP1000 DC. The purpose is to determine if the DAMSAM and Shield-APR codes together include enough conservatism to account for not including the buildup of daughter progeny downstream of the RCS. If the DAMSAM and Shield-APR codes together result in higher source terms than the Westinghouse codes, this may be used to assist in providing justification for not considering daughter product buildup in the APR1400 design. The response also indicated that the final response will provide information on the conservatisms with the DAMSAM/Shield-APR code methodology. Therefore, the final response is expected to not only provide the results of the benchmark calculations but also to provide any specific information on how the DAMSAM and Shield-APR code methodology is conservative. The specific information regarding why the DAMSAM and Shield-APR codes are conservative is necessary to resolve this issue. The staff notes that this issue could potentially impact all of the normal operation source terms to some extent. **RAI 343-8420, Question 12.02-23, is being tracked as an Open Item**, until the final response is submitted.

The staff identified numerous significant radiation sources for which the source terms were not included in the application. These include source terms for the pressurizer, refueling water storage tank, and hold-up volume tank. The staff **issued RAI 13-7856, Question 12.02-2 (ML15142A607)**, requesting that the applicant provide source terms for these sources. In addition, in RAI 13-7856, Question 12.02-2, the staff requested that the applicant provide a source term for gases within the volume control tank. Finally, for many of the sources provided in DCD Tier 2, Section 12.2, the applicant did not provide any of the methods, models, and assumptions used as the basis for the source terms. Therefore, consistent with the standard review plan, the staff requested that the applicant provide this information in the DCD for the CVCS tanks and boric acid storage tank (BAST), as well as other sources where the methods, models, and assumptions had not been provided.

In its response to **RAI 13-7856, Question 12.02-2 (initial ML15202A672, revised ML15258A675)**, the applicant provided radioactive source terms for the pressurizer and refueling water storage tank, including the methods, models, and assumptions made in developing the source terms, as well as the proposed changes to the DCD Tier 2, Section 12.2. The applicant also explained that the hold-up volume tank is designed to collect water during an accident and does not contain radioactive material during normal operating conditions. Therefore, there is no source term during normal operation for this tank. In addition, the applicant provided the methods, models, and assumptions for calculating the vapor source term for the volume control tank and other tanks and updated the tank source terms accordingly. Finally, the applicant provided the methods, models, and assumptions for calculating the source terms of CVCS components and provided source term information for the SFP demineralizer and filter and updated the DCD with this information.

However, the applicant's response indicated that many of the liquid tank source terms were based on the tanks only being filled to a fraction of their total volume. For example, the proposed addition to DCD Tier 2, Table 12.2-25, indicated that the holdup tank source term was based on the tank only being filled to 12.5 percent of its total capacity and that the BAST was only filled to 50 percent of its total capacity. Similar issues were identified for the reactor makeup water tank (RMWT), reactor drain tank, equipment drain tank, and in-containment refueling water storage tank (IRWST). Since the SRP indicates that normal operation shielding and zoning should be based on the maximum source term, the **staff closed RAI 13-7856, Question 12.02-2, as an unresolved item**. Additionally, the staff issued **RAI 343-8420, Question 12.02-22 (ML15356A016)**, requesting that the applicant revise the source term calculations and associated shielding and zoning for these tanks and to update the DCD appropriately. In addition, in RAI 343-8420, Question 12.02-22, the staff requested that the applicant provide additional editorial corrections and clarifications. Finally, RAI 343-8420, Question 12.02-22, also includes questions associated with outdoor tanks, which are also discussed within the context of the review of RAI 13-7856, Question 12.02-3, below. The applicant's response to RAI 343-8420, Question 12.02-22, is discussed below, following the discussion regarding RAI 13-7856, Question 12.02-3.

In Revision 2 of its response to **RAI 13-7856, Question 12.02-2, (ML16279A528)**, the applicant updated the activities in the holdup tank, BAST, and IRWST as a result of the applicant's response to RAI 343-8420, Question 12.02-22 (below). However, the revised source terms had very low concentrations of Ba-137m. Ba-137m has a short half-life and is generated from the decay of Cs-137. It is one of the most significant radionuclides from a radiation shielding and zoning perspective. Since the Shield-APR code does not account for the decay of the parent radionuclides, the applicant had assumed that the activity of Ba-137m was the same as Cs-137 for all other sources. It is acceptable to assume that Cs-137 and Ba-137m have the same activity because Ba-137m would be expected to be essentially the same activity as Cs-137 because its half-life is much shorter than Cs-137. Since the revised source terms do not accurately consider Ba-137m, they are incomplete. In addition, while Revision 2 of the response modified the source terms of the holdup tank, BAST, and IRWST; the revised response did not modify any of the dose rate information in Table 5, "Contact Dose Rates for the Two Cases," of the response, which was based on the original source terms and is misleading. These issues are expected to be addressed in a future revision to the response to **RAI 13-7856, Question 12.02-2. This is now being tracked as an Open Item**, until these issues are resolved.

In reviewing the application, the staff also determined that there were additional inadequacies with the information provided for the holdup tank, RMWT, and BAST, which are located outdoors. Specifically, the application did not provide the dimensions or parameters used in the shielding analysis for these tanks or provide shield thicknesses or radiation zones for the tanks. In addition, while the applicant indicated that there will be administrative controls in place to prevent personnel from occupying the immediate vicinity of the tanks, the applicant did not describe the administrative controls or provide a COL information item to ensure that the COL applicant will provide these administrative controls. Finally, while the holdup tank is a significant radiation source, the source term is lower than what is provided for similar tanks in other new reactor design applications. Therefore, the staff requested that the applicant provide more information on how the holdup tank source term was calculated. The staff requested the applicant to provide all of the above information in **RAI 13-7856, Question 12.02-3 (ML15142A607)**.

In its response to **RAI 13-7856, Question 12.02-3 (initial ML15202A672, revised ML15258A675)**, the applicant provided source term dimensions and parameters for the holdup tank, RMWT and BAST. In addition, the applicant indicated that each tank is surrounded by a dike with a key lock to the entrance. The applicant also indicated that the holdup tank and BAST are surrounded with a layer of concrete shielding around the tanks, from bottom to top, with no gap between the tank surface and concrete wall and that all of these tanks are designed to maintain dose rates to less than 0.25 mrem/hour (2.5 μSv/hr) at their surface (for the holdup tank and BAST, this is the surface of the concrete), based on the assumed 0.25 percent fuel failure source terms.

However, RG 8.8 indicates that station features and design should, to the extent practicable, permit inspections to be accomplished expeditiously and with minimal exposure to personnel. It is unclear how inspections will be performed while minimizing dose to workers. In addition, while the holdup tank and BAST are shielded on all vertical sides, there is no shielding included for the tops of the tanks and it is unclear if doses on the auxiliary building roof or other nearby elevated areas could be effected by the tanks. Finally, in **RAI 13-7856, Question 12.02-3 (ML15142A607)**, the staff requested that the applicant provide the methods, models, and assumptions used for calculating the source term for the holdup tank. While the applicant indicated that the tank activities are the sum of normal operations, shutdown boration operation, drain operation for reactor vessel removal, and volume added during contraction cooldown, the applicant did not provide the radionuclide concentration or volume of each of these inputs to the holdup tank source term. As a result, the **staff closed Question 12.02-3 as an unresolved item** and requested that the applicant provide this information in **RAI 343-8420, Question 12.02-22 (ML15356A016)**. RAI 343-8420, Question 12.02-22, also includes follow-up questions to RAI 13-7856, Question 12.02-2, as discussed above.

In its response to **RAI 434-8420, Question 12.02-22 (ML16279A524)**, the applicant described how the source terms were calculated for the liquid tanks. For the holdup tank, the activity in the liquid volume of the tank is based on 62 percent of the tanks total volume (20 percent from normal power operation volume, which is from the RCS, filtered through the CVCS system and 42 percent from shutdown, refueling, and startup operations). The total activity based on 62 percent volume is then concentrated to 12.5 percent of the tanks total volume for the shielding calculation. The vapor phase was calculated assuming 5 percent of the tank liquid volume (95 percent of the volume is based on the gas phase), which maximizes the gas phase source term. The total activity of the vapor phase was calculated based on the RCS fluid processed through the CVCS system, with consideration for the operation of the gas stripper. The fluid from the CVCS system is the most radioactive fluid and would result in the highest source term. The gaseous source terms are based on the activity in the CVCS fluids and Henry's Law, except for radionuclides that are not in equilibrium, as described in the response. For radionuclides that are not in equilibrium (for the holdup tank these radionuclides are krypton-85 (Kr-85), xenon-131m (Xe-131m), and Xe-133), the applicant calculated the gaseous source term based on a partition factor and radioactive decay, assuming the operation of the tank for one reactor cycle. The activity in the gaseous tank volume of 95 percent is then condensed to 87.5 percent for the shielding calculation. While in DCD Table 12.2-25 (as updated in its response to RAI 13-7856, Question 12.02-3), the applicant indicated that the shielding calculation for the holdup tank is based on 12.5 percent liquid volume and 87.5 percent volume for the vapor phase, the activities in the liquid and gaseous phases are actually based on 62 percent volume and 95 percent volume, respectively. Therefore, the total activity in the tank is based on a larger total volume than the tanks' actual capacity. In addition, the gaseous source term contributes about 90 percent of the total dose of the tank. Therefore, it is more conservative to assume the tank volume is dominated by gas than liquid (increasing the liquid source term and lessoning the gaseous by the same amount would result in a smaller overall dose rate). See the staff's evaluation of the response to RAI 308-8339, Question 12.02-19, for additional information related to the acceptability of the holdup tank source term. As a result of the responses to RAI 308-8339, Question 12.02-19 and RAI 343-8420, Question 12.02-22, the staff determined that the applicant's methodology for calculating the holdup tank shielding and zoning is acceptable.

Finally, the applicant proposed a correction (slight increase) to the holdup tank source term in Revision 2 of the response to RAI 13-7856, Question 12.02-2 (see above). The basis for the holdup tank source term and components downstream of the holdup tank is based largely on the high decontamination factor for cesium (Cs) and rubidium (Rb) in the pre-holdup ion exchanger (see the applicant's response to **RAI 308-8339, Question 12.02-19 (ML16272A470)**) and the assumption that the gas stripper will be continually operated when radioactivity levels in the CVCS are high. Therefore, during the teleconference on August 23, 2016, the applicant agreed to update DCD Tier 1 to include a statement indicating that the gas stripper would be operated as necessary to meet Zone 1 (0.25 mrem/hr or 2.5μSv/hr) dose rates in outside areas near the holdup tank (when considering holdup tank shielding) and as necessary to maintain offsite doses to less than the values in 40 CFR 190, "Environmental Radiation Protection Standards for Nuclear Power Plants," as required by 10 CFR 20.1301(e). The staff notes that, since there is no shielding on the top of the tank, the dose on the auxiliary building roof is up to 2.5 mrem/hr (25 μSv/hr) from the tanks that are over 50-feet (15.2 meters) away. Since the dose rate could be up to 2.5 mrem/hour over 50 feet away from the tanks, there is a potential for a noticeable dose contribution offsite due to these tanks, depending on the site layout and the proximity of the site boundary to these tanks. This raises a potential concern of compliance with 40 CFR Part 190.

However, in its response to **RAI 343-8420, Question 12.02-22 (ML16279A524)**, the applicant proposed updating DCD Tier 1, Section 2.4.6, "Chemical and Volume Control System," to only state that the gas stripper in CVCS system is operated as necessary to ensure that the dose rate outside the holdup tank remains less than the Zone 1 criteria. In addition, there is no information in DCD Tier 2 or any COL item to specify the site boundary dose from the outdoor tanks in accordance with 40 CFR 190 or to specify that the COL applicant will provide this information. Because there is no information on 40 CFR 190 compliance related to the outdoor tanks, the applicant's response is incomplete. This issue is to be addressed in a future update to the response to Question 12.02-22. **This is being tracked as an Open Item associated with the review of Question 12.02-22**.

In its response to Question 12.02-22, the applicant also explained that the RMWT, reactor drain tank, and equipment drain tank source terms and shielding and zoning calculations were calculated using a similar methodology to that used for the holdup tank. The staff reviewed the methodology used to calculate the source terms for these tanks and determined that they are acceptable. However, the BAST and IRWST do not contain a significant gaseous source term and were still not based on the maximum tank volume. Therefore, in its response, the applicant proposed updating the tank volumes to their high level alarms, which are near the full capacity of the tanks (95 percent for BAST and 87 percent for the IRWST). The applicant indicated that the source terms for these tanks would be updated in Revision 2 to the response to Question 12.02-2 (see above). Due to these changes, the applicant also increased the source term for the boric acid filter in its response to Question 12.02-22. This change is in accordance with the revised holdup tank and BAST source terms and is, therefore, acceptable. The applicant also included the minimum required shielding thicknesses for the BAST and the holdup tank in DCD Tier 2, Table 12.3-4, which is acceptable.

In its response to RAI 343-8420, Question 12.02-22, the applicant also provided detailed information on the location of the holdup tank and BAST and the dose rate on the auxiliary building roof from these sources. Specifically, the applicant indicated that the tanks are over 50-feet (15.2 meters) away from the auxiliary building and that the dose rate on the auxiliary building roof could reach radiation Zone 2 (up to 2.5 mrem/hr (25 μSv/hr)) from the tanks. This calculation does not consider the shielding of a tank roof, which would be expected to possibly provide minimal shielding. However, the applicant did not include any radiation zoning information for the auxiliary building roof in DCD Tier 2, Chapter 12. In addition, as discussed above, the applicant did not provide any information on the potential dose contribution at the site boundary from these tanks or any COL item to specify that the COL applicant will provide this information. **This is being tracked as an open item associated with the review of Question 12.02-22.**

Since the holdup tank and BAST are surrounded by concrete from the bottom to the top of the tank, the staff requested that the applicant provide information explaining how this design meets the criteria in RG 8.8, which states that station features and design should, to the extent practicable, permit inspections to be accomplished expeditiously and with minimal exposure of personnel. It is unclear how the inspection of the tank would be performed and leaks identified. In Part 7 of RAI 343-8420, Question 12.02-22, the staff requested that the applicant provide information on this issue. In its response to Question 12.02-22 (ML16279A524), the applicant specified that this information would be provided at a later date. **This is being tracked as an open item associated with the review of Question 12.02-22.**

Likewise, in Part 8 of Question 12.02-22, the staff had requested that the applicant clarify various apparent inconsistencies in its response to RAI 13-7856, Question 12.02-2 (ML16279A528) and associated information in the DCD. The applicant also specified that this information will be provided at a later date. **This is being tracked as an open item associated with the review of Question 12.02-22.**

As a result of the remaining open items discussed above, **RAI 343-8420, Question 12.02-22 is being tracked as an Open Item**, until the applicant provides a response to all parts of the question and the staff completes its review.

In reviewing radiation sources outside containment, the staff also identified that the application did not provide source term information for the SFP cleanup demineralizer, SG blow down flash tank, and waste storage drum area. Also, while source term information was provided for the following sources, the applicant did not provide any dimensions or parameters in DCD Tier 2, Table 12.2-25, for the following components;

- Seal injection filter
- Reactor drain filter
- Boric acid filter
- Purification filter
- Reactor makeup water filter
- Concentrate heater
- Concentrate cooler
- Flash tank
- Vapor separator
- Concentrate pump
- Concentrate transfer pump
- SG blowdown mixed-bed
- Blowdown pre-filter
- Blowdown post-filter
- CPS cation bed
- CPS mixed bed
- Equipment waste tank
- Monitor tank
- Reverse osmosis

In addition, while DCD Tier 2, Table 12.2-25 provides dimensions and parameters for the liquid radwaste system ion exchanger, DCD Tier 2, Section 11.2 "Liquid Waste Management System," and Table 12.2-21 "Liquid Radwaste System Component Source Terms (Bq)" indicate that the LWMS includes a cation bed and two mixed bed ion exchangers (identified as mixed bed 1, and mixed bed 2 ion exchangers). It is unclear to which of these components the provided dimensions and parameters would apply. Table 12.2-25 should include the dimensions for all of the sources terms provided in Chapter 12. Therefore, the staff issued **RAI 103-7998, Question 12.02-12 (ML15205A401)**, requesting that the applicant provide this information.

In its response to **RAI 103-7998, Question 12.02-12 (ML15245A493)**, the applicant provided all of the requested information. However, the applicant also indicated that, for determining plant shielding and zoning, all components in the boric acid concentrator package (concentrate heater, concentrate cooler, flash tank, vapor separator, concentrate pump, and concentrate transfer pump) were considered to be contained within the flash tank, since all of the components are located within the same cubicle. This is considered to be a conservative assumption because this concentrates all the radioactive material into one area, where actually the source term would be more dispersed, lessening the dose rates to the adjacent rooms. However, the applicant did not include this explanation in the DCD. Therefore, the **staff closed Question 12.02-12 as an unresolved item**, and **issued RAI 321-8353, Question 12.02-21 (ML15329A236)**, requesting that the applicant explain why this information is not in the DCD.

In its response to **RAI 321-8353, Question 12.02-21 (ML16029A025)**, the applicant proposed updating DCD Tier 2, Subsection 12.2.1.1.5.1.e, "Boric acid concentrator," to specify that the components of the boric acid concentrator are conservatively modeled by summing all of the individual source terms to determine the minimum shield wall thicknesses for the boric acid concentrator room. The source term is modelled as a cylinder, as described in DCD Tier 2, Table 12.2-25. As discussed above, this approach is conservative and is acceptable. However, the staff was unable to match the applicant's source term for the boric acid concentrator and BAST using the holdup tank source term. In its response to RAI 343-8420, Question 12.02-22 (discussed above), Part 3, the applicant clarified that the source term for the boric acid concentrator and BAST were based on processing holdup tank fluid, assuming that only fluid in the holdup tank is based on activity from normal operation (not considering shutdown operation), while the holdup tank source term considers shutdown operation and was concentrated to 12.5 percent of the tank volume for shielding purposes. Considering that all of the radioactivity comes from normal operation is conservative because it results in a higher radionuclide concentration than if it is mixed with shutdown fluid, which would have a lower radioactivity concentration. When considering only the activity during normal operation and the proper volume and concentration factor of 100, the staff is able to verify the activity in the boric acid concentrator and BAST. As a result, the boric acid concentrator and BAST source terms are acceptable (when also considering the applicant's response to RAI 343-8420, Question 12.02-22). Therefore, **Question 12.02-21, is being tracked as a confirmatory item**, pending the proposed DCD revisions.

Also regarding DCD Tier 2, Table 12.2-25, the staff noted that the shutdown cooling heat exchanger vapor phase density is listed as 0.453 grams per cubic centimeter, (28.3 pounds per cubic foot), which is a much higher density than would be expected for water vapor. The staff issued **RAI 23-7929, Question 12.02-9 (ML15174A324)**, requesting that the applicant clarify or justify this value. In its response to **RAI 23-7929, Question 12.02-9 (ML15223B087)**, the applicant indicated that there was a typographical error in DCD Table 12.2-25, when referring to the water vapor density and that the density of 0.453 grams per cubic centimeter is the partial density of steel in the heat exchanger, which makes up 6 percent of the material in the heat exchanger. The staff's review concluded that 6 percent of the density of steel is approximately 0.453 grams per cubic centimeter. The applicant proposed updating DCD Table 12.2-25, to specify that the density of 0.453 grams per cubic centimeter, for the shutdown cooling heat exchanger represents 6 percent of the density of steel. Therefore, **the applicant's response to Question 12.02-9, is acceptable and is being tracked as a confirmatory** item, pending the correction of the typographical error in the DCD.

In reviewing the gaseous radwaste system source terms, the staff identified several missing source terms. The missing sources were the header drain tank, guard bed, waste gas dryer, and gaseous radwaste system high efficiency particulate air (HEPA) filter. In addition, the dimensions and parameters for these components and the delay beds were not provided in DCD Tier 2, Table 12.2-25. The staff requested additional information on the assumptions used in calculating the gaseous radwaste system source terms. Therefore, the staff issued **RAI 103-7998, Question 12.02-11 (ML15205A401)**, requesting that the applicant provide this information.

In its response to **RAI 103-7998, Question 12.02-11 (ML15245A493)**, the applicant provided the missing source terms information, including the associated dimensions and parameters. The applicant also provided more detailed information in the DCD regarding how the source terms for gaseous radwaste system were calculated. However, the 0.25 percent fuel defect waste gas dryer source term was significantly lower than that for the 1 percent failed fuel percent source term without any justification for this difference. Furthermore, the buildups of daughter nuclides were not included in the source terms and some daughters could be significant for the shielding of gaseous radwaste system components, especially Rb-88. Finally, additional descriptions were needed in the DCD regarding some of the information added, and there were inconsistencies in the dimensions provided for the header drain tank in the proposed DCD Table 12.2-25, and the volume provided in DCD Tier 2, Table 11.3-4, "GRS Major

Equipment Design Information.” Therefore, the **staff closed Question 12.02-11**, as an unresolved item and issued **RAI 343-8420, Question 12.02-25 (ML15356A016),** requesting that the applicant provide the requested information and to resolve these issues. Therefore, **RAI 343-8420, Question 12.02-25, is being tracked as an Open Item**.

In addition to the missing source terms and missing source parameters requested in the above RAIs, the applicant indicated that the decontamination factors for waste treatment components are based on NUREG-0017, “Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors,” Revision 1. The decontamination factors in NUREG-0017 are acceptable for use in estimating decontamination factors for components, because RG 1.112, “Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors” (which is referenced in Chapter 12 of the SRP as a basis for developing source terms) references NUREG-0017, unless actual plant design is known to provide different decontamination factors. However, several decontamination factors for ion exchangers did not match the values provided in NUREG-0017. For example, the CVCS pre-holdup mixed bed ion exchanger decontamination factor is listed as 100 for Cs and Rb, even though NUREG-0017 listed the decontamination factor as 2. In addition, the DCD indicates that removal efficiency of yttrium in all CVCS ion exchangers was assumed to be 1, which is also inconsistent with NUREG-0017. Therefore, the staff issued **RAI 15-7896, Question 12.02-4 (ML15142A609),** requesting that the applicant provide the basis for decontamination factors different than what is provided in NUREG-0017 or revise the DCD, as appropriate.

In its response to **RAI 15-7896, Question 12.02-4 (ML15204A727)**, the applicant listed all decontamination factors assumed for the CVCS system, SG blowdown system, LWMS, and condensate polishing system, in developing the DCD Tier 2, Section 12.2 source terms. Most decontamination factors were consistent with NUREG-0017 and the guidance in SRP Section 12.2 and are acceptable. However, several of the decontamination factors provided were inconsistent with NUREG-0017, with no basis provided for the values used, including decontamination factors used for the pre-holdup ion exchanger. In addition, the applicant assumed a decontamination factor of 1 for yttrium, for all CVCS system components, based on WASH-1258, ("Proposed Rule Making Action, Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion "As Low As Practicable" for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents," July 1973). WASH-1258 pre-dates NUREG-0017 and is not referenced in SRP Chapter 12 as a basis for developing source terms. Therefore, use of the information in WASH-1258 as an alternative to NUREG-0017 must be accompanied with proper justification for its use. Hence, the **staff closed Question 12.02-4 as an unresolved item** and issued **RAI 308-8339, Question 12.02-19 (ML15320A154)**, requesting that the applicant provide additional justification for the decontamination factor values which are different than those specified in NUREG-0017.

In its response to **Question 12.02-19 (ML16272A470)**, the applicant provided justification for the decontamination factors used. For the decontamination factors of cesium and rubidium of 100 for the pre-holdup ion exchanger, the applicant indicated that there is a history of Combustion Engineering plants using a decontamination factor of 100 for the ion exchanger proceeding the holdup tank. The applicant also indicated that there is substantial data dating back to the 1970s showing that a decontamination factor of greater than 100 can be maintained for an extended period (one operating cycle or greater), with the use of a 3:1 cation to anion bed resin volume ratio ion exchanger. The response included references and notes from numerous documents to support this claim. In addition, since the decontamination factor of 100 is significantly greater than the factor of 2 recommended in NUREG-0017 and because this decontamination factor is necessary to ensure the radiation shielding and zoning in outside areas is as described in the DCD, the applicant proposed updating DCD Tier 1, Section 2.4.6.1 and Tier 2, Section 12.2.1.1.5.1, “Chemical and Volume Control System,” to specify that the pre-holdup ion exchanger has a 3 to 1 cation to anion bed resin volume ratio and provides a minimum cesium decontamination factor of 100. This ensures that the pre-holdup ion exchanger will be designed to maintain a decontamination factor of 100 for cesium and rubidium for the life of the facility and is acceptable.

Also, in its response to Question 12.02-19, the applicant indicated that there is a history of assuming a decontamination factor for yttrium of 1 in Westinghouse plants and that yttrium is not specifically called out in NUREG-0017. Since the yttrium isotopes are mostly beta emitters, without any associated significant gammas in large quantities, yttrium would be negligible in determining the radiation shielding and zoning for plant components. Likewise, since yttrium is not being collected by any of the ion exchangers the amount of yttrium assumed to be in the plant piping is maximized. Accordingly, the quantity in the airborne activity concentrations are maximized. Based on the above, the staff concludes that the use of a decontamination factor of 1 for yttrium is acceptable, for the purposes of the information provided in DCD Tier 2, Chapter 12.

Finally, in its response to Question 12.02-19, the applicant also clarified that although there are two SG blowdown ion exchangers that are normally operated in series, the design provides the flexibility to operate the ion exchangers individually. Therefore, the source terms for both ion exchangers are conservatively based on the operation of one ion exchanger. This maximizes the source terms and shielding for the ion exchangers and is acceptable. The applicant also proposed updating DCD Tier 2, Table 11.1-5, “Assumptions Used in Determining Secondary System Activities,” to correct the decontamination factor for the blowdown demineralizer for cesium and rubidium. The value is being corrected from 100 to 10. The staff finds that this change is consistent with that cited in NUREG-0017 and is, therefore, acceptable. The applicant also proposed to add a note to DCD Tier 2, Table 11.1-5, specifying that in determining the source term for secondary systems, only one SG is assumed to be operating. The staff finds this note acceptable.

As a result of the above, the response to **Question 12.02-19, is being tracked as a confirmatory** item, pending the incorporation of the proposed DCD updates.

Regarding the SG blowdown source terms, DCD Tier 2, Subsection 12.2.1.1.5.2, “Steam Generator Blowdown System,” indicates that the blowdown rate for calculating SG blowdown system source terms is assumed to be 0.2 percent of the maximum steaming rate, while DCD Tier 2, Subsection 10.4.8.2.3, “System Operation,” indicates that the blow down rate may be as high as 1 percent of the SGs maximum steaming rate until water quality is within the normal limits. Since SRP Section 12.3-12.4 specifies that the radiation zoning and shielding are to be based on the maximum source term that would be expected during normal operation and anticipated operational occurrences, the staff requested that the applicant justify using 0.2 percent of the maximum steaming rate in **RAI 15-7896, Question 12.02-5 (ML15142A609)**. Also in RAI 15-7896, Question 12.02-5, the staff requested that the applicant correct apparent discrepancies and provide additional clarifications for the assumptions used in calculating condensate polishing system source terms.

In its response to **RAI 15-7896, Question 12.02-5 (ML15204A727)**, the applicant indicated that 1 percent of the maximum steaming rate was assumed in calculating SG blowdown system sources and that the value of 0.2 percent was a typographical error and proposed to correct the

DCD. The staff reviewed the applicant's calculations provided during a Chapter 12 source term audit and determined that the value of 1 percent was used in the source term calculations, which is acceptable. In addition, the applicant provided additional clarifications on how the condensate polishing system source terms were calculated and proposed editorial corrections to DCD Tier 2, Subsection 12.2.1.1.5.3, "Condensate Polishing System," making it consistent with DCD Chapter 10. The staff determined that the proposed DCD changes are acceptable because the source term is based on the maximum steaming rate and the editorial errors were corrected. Therefore, **RAI 15-7896, Question 12.02-5, is being tracked as a confirmatory item**, pending the incorporation of the proposed DCD revisions.

As a result of the source term audit and subsequent staff review, the staff also determined that additional information was needed in the DCD to discuss the replacement frequencies assumed in developing the source terms for filters and demineralizers in the SG blowdown and spent pool cleaning system. In addition, the assumptions for developing the source term for the SWMS were not included in the DCD. The staff issued **RAI 207-8247, Question 12.02-15 (ML15295A505)**, requesting the applicant to provide this information.

In **Revision 2 of its response to Question 12.02-15 (ML16197A393)**, the applicant proposed updating DCD Tier 2, Section 12.2.1.1.5.2, to state that the SG blowdown system pre-filter, post-filters, and mixed beds are calculated based on radioactive crud and nuclide buildup at the end of 6 months of processing. The applicant also proposed updating DCD Tier 2, Subsection 12.2.1.1.5.3 to specify that the condensate polishing system (CPS) cation and mixed beds (7 of each) are based on radioactivity buildup at the end of their corresponding processing cycles, the nuclide accumulation times for the cation beds are based on 3 days of processing and 30 days for the mixed beds. DCD Tier 2, Subsection 12.2.1.1.3, "Secondary-Side Systems," specifies that the source term for secondary-side systems is based on an assumed SG tube leakage rate of 0.3 gallons per minute (1,635 liters per day) in each SG, or a total of 0.6 gallons per minute (3,270 liters per day). This leakage far exceeds the TS limit of 150 gallons per day (568 liters per day) through one SG specified in TS Section 3.4.12, "RCS Operational Leakage." The likelihood of operating at the TS limit of primary to secondary leakage with 0.25 percent failed fuel, for an extended period of time, is very low. Since the applicant assumed a primary to secondary leak rate far exceeding the limit by TS, the staff accepts the short processing times for these components. In addition, the short processing time is consistent with DCD Tier 2, Subsection 10.4.6.2.4, "Design Features for Minimization of Contamination," which states that the CPS cation and mixed-bed demineralizer columns are designed with periodic resin replacements to prevent the buildup of contaminants, thus minimizing the potential of accumulation of significant amounts of contamination and waste generation. As specified in DCD Tier 2, Section 10.4.6.2.3, spent CPS resin is transferred to the spent resin holding tanks, where temporary shielding can be installed, if required. Radioactive resin is transferred from the spent resin holding tank to the radwaste treatment area for waste management. Therefore, the source term for the CPS system is acceptable.

In Revision 2 of its response to Question 12.02-15 (ML16197A393), the applicant also proposed updating DCD Tier 2, Section 12.2.1.2.4 to specify that the source terms for the SFP demineralizers and filters are integrated over the cleanup time for normal operation and are determined to be at maximum at about 265 hours for the filters and 290 hours for the demineralizers, after which time the source terms decrease due to decay of short half-life nuclides. This is appropriate, because as the SFP cleanup system cleans the SFP during refueling operation, the pool will continue to be cleaned and less radioactivity will enter the cleaning system. Therefore, the filters and demineralizers will reach a maximum value, after which time the source terms will decrease, because of the continually cleaner SFP water being processed. The staff reviewed the applicant's calculations for the SFP filters and demineralizers and determined that they are acceptable.

Finally, the applicant also proposed updating DCD Tier 2, Subsection 12.2.1.4 to describe the basis for the source terms for the SWMS provided in DCD Table 12.2-22. The applicant specified that the source term for the spent resin long-term storage tank are calculated based on 10 years of cumulative CVCS resin. In addition, the applicant specified that the source terms for the low-activity spent resin storage tank demineralizers are based on a processing time of 1 year.

However, confirmatory staff calculations showed that the source term for the low-activity spent resin storage tank was actually based on a source term that exceeded 1 year of LWMS storage. In addition, while the source term for the CPS demineralizers was acceptable, the applicant did not provide information on shielding for the CPS demineralizers, including their locations and minimum shielding thicknesses in DCD Tier 2, Table 12.3-4.

In **Revision 1 of its response to Question 12.02-15 (ML16124B180)**, the applicant clarified that the source term for the low-activity spent resin tank in DCD Tier 2, Table 12.2-22 was calculated using the source term for the LWMS spent resin and multiplying by a factor of 3. Since the low-activity spent resin tank is filled with resin from the LWMS, SG blowdown system, and SFP cleaning system and the source terms for the LWMS are highest, it is conservative to assume that the tank is filled with three times the maximum LWMS resin. Therefore, the staff determined that the source term is acceptable.

In Revision 2 of its response to Question 12.02-15 (ML16197A393), the applicant provided numerous additional information and clarifications, including showing the location of the CPS demineralizer area in DCD Tier 2, Figure 12.3-17, "Radiation Zones (Normal) Turbine Building El. 73'-0"," and providing new COL Information Item 10.4(12). COL Information Item 10.4(12) specifies that "[T]he COL applicant is responsible for provisions of temporary shielding, if required, and mobile equipment, including spent resin fill-head for packaging of the contaminated spent resin, provisions of temporary storage, and shipment of packaged contaminated CPS spent resin for off-site treatment and/or disposal." It is acceptable for the COL applicant to provide mobile equipment and temporary shielding, as required, since mobile equipment and temporary shielding may only be necessary, if unfavorable operating conditions occur. The response also indicates that the CPS demineralizers are located in a shielded area on the 73'-0" elevation of the turbine building. The CPS demineralizer is in a Zone 2 area, which is already at a low dose rate to provide unlimited access to workers (40 hrs. per week). The additional shielding will ensure that the dose rates to areas outside the CPS demineralizer area is a Zone 1 dose limit, as indicated on the proposed revision to DCD Tier 2, Figure 12.3-17. Therefore, the response is acceptable and Revision 2 of the response to **Question 12.02-15 is being tracked as a confirmatory item**, pending the proposed DCD revisions.

Also, following the source term audit and the review of DCD information, the staff determined that the source term provided for the monitor tanks was insufficient because it did not properly account for the potential for filling a monitor tank with fluid that was processed through the LWMS, with an initial concentration of RCS fluid. In addition, the information in the DCD describing the assumptions used for calculating LWMS source terms was insufficient. Specifically, while the applicant's calculations show that LWMS source terms are based on processing the 0.25 percent failed fuel RCS source term diluted to 44 percent of the RCS concentration based on some of the input paths coming from waste tanks which have had radionuclide decay and sources which have already been processed (such as through the

CVCS system), the DCD implied that the LWMS source terms are based on processing 100 percent RCS concentration at 0.25 percent failed fuel. The staff issued **RAI 207-8247, Question 12.02-17 (ML15295A505)**, requesting the applicant provide additional information regarding this discrepancy.

In its response to **RAI 207-8247, Question 12.02-17 (ML16028A182)**, the applicant proposed updating the monitor tank source term to base the source term on the LWMS processing RCS fluid with an activity based on an assumed 0.25 percent failed fuel as the RCS source term. It is necessary to use 100 percent of the RCS source term, processed through the LWMS, for the monitor tanks because the monitor tanks source term would likely be based on fluid processed over a short period of time (i.e., previous hours or days). It is reasonable that this could be fluid at 100 percent RCS concentration. Therefore, it is appropriate to use this assumption for the monitor tanks. This is in accordance with the guidance in SRP Chapter 12. The staff reviewed the source term and found the newly calculated activities to be consistent with using 100 percent RCS fluid and is, therefore, acceptable.

In its response to Question 12.02-17, the applicant also proposed DCD changes to update the DCD with the revised source term information and to explain the assumptions used in calculating the source terms for LWMS components. As indicated above, the source terms for LWMS components other than the monitor tanks are based on processing fluid with radionuclide concentrations of 44 percent of the RCS source term (assuming 0.25 percent failed fuel). This is acceptable for LWMS source terms other than the monitor tanks because these source terms are based on the buildup of radionuclides over a period of a year. Over a period of a year, the LWMS would process radioactive waste with various different concentrations. As a result, the combination of input pathways resulting in an assumed average concentration at 44 percent of the RCS concentration with 0.25 percent failed fuel over a year, is acceptable. Finally, in its response, the applicant indicated that it performed confirmatory calculations for the shielding and zoning around the monitor tank area and confirmed that none of the shield wall thicknesses or radiation zones need to be changed as a result of the source term change. The staff's review of the shielding and zoning for the monitor tank and surrounding areas, using simplified assumptions based on some of the most radiologically significant radionuclides, confirmed results consistent with the applicant's calculations of the shielding and zoning for the monitor tanks. Therefore, **Question 12.02-17 is being tracked as a confirmatory item**, pending the proposed DCD changes.

During the source term audit (ML16119A083), the staff also verified the acceptability of numerous other sources including the CVCS purification ion exchanger source term, which is based on each purification ion exchanger being used for both regular radionuclide removal from the RCS fluid and for lithium removal. This assumption is conservative because it accounts for extra buildup in the ion exchanger than would actually be expected during one operating cycle.

The staff reviewed the APR1400 contained sources in accordance with SRP Section 12.2, “Radiation Sources,” Revision 4. Upon complete resolution of the open items, the staff will determine if the design of the APR1400 meets Commission requirements.

12.2.4.2 Airborne Radioactive Material Sources

The guidance contained in RG 1.206, “Combined License Applications for Nuclear Power Plants,” states that the applicant should describe in Section 12.2 of the DCD Tier 2, "those airborne radioactive sources in the plant that are considered when designing the ventilation systems and in specifying appropriate monitoring systems." This description should include a tabulation of the calculated concentrations of "radioactive material by nuclide expected during normal operation, AOO, and accident conditions for equipment cubicles, corridors, and operating areas normally occupied by operating personnel." It should also include models and parameters for the calculations. In DCD Tier 2, Section 12.2.2, "Airborne Radioactive Material Sources," the applicant described the sources of airborne radioactivity for the reactor design. The DCD indicates that airborne radioactive material is introduced into the plant atmosphere through the leakage of radioactive fluids from equipment and the evaporation of radioactively contaminated water.

While the APR1400 is designed to minimize the potential for leakage of radioactive fluids, the applicant assumed that pump seals, valve stems, and other piping connections would leak based on engineering margin from design specifications to estimate airborne activity concentrations for the main cubicles in the plant. To do this, the applicant determined the concentration of radioactive fluid (assuming 0.25 percent failed fuel) and the potential leak rate within each cubicle, based on equipment design specifications, in each of the main cubicles (except for the fuel handling area, where the source is from the evaporation of the SFP and not the leaked RCS fluid). Using the leak rates, radionuclide concentrations, room volumes, and partition coefficients for different radionuclides, the applicant calculated the amount of airborne radioactive material in each cubicle and determined the minimum ventilation system flow rate for each cubicle to ensure that airborne activity concentrations are minimized to the extent practicable and are below the derived air concentration (DAC) limits prescribed in Table 1, "Occupational Values," of 10 CFR Part 20, Appendix B, "Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage."

To determine the minimum required ventilation flow rates to reach the determined concentration, the applicant used the inputs described above and the equation provided in DCD Tier 2, Subsection 12.2.2.3. This equation is an acceptable method for estimating airborne activity concentrations and has been used to estimate airborne activity concentration in other new reactor designs. The specific inputs to this equation for each cubicle are provided in DCD Tier 2, Table 12.2-23, "Airborne Radioactivity Concentrations," and Table 12.2-26, "Assumptions and Parameters, Used in Airborne Source Term Calculations." However, in the description of airborne activity sources, the application did not include enough information for the staff to duplicate the applicant's results. The airborne activity levels included consideration of only the isotopes, krypton (Kr), xenon (Xe), bromine (Br), iodine (I) isotopes, as well as tritium, and the application was unclear why these were the only radionuclides included.

The staff issued **RAI 23-7929, Question 12.02-7 (ML15174A324)**, requesting that the applicant provide additional information on the calculation of airborne activity source terms. Specifically, the staff requested that the applicant provide a walkthrough for performing calculations for airborne activity levels in different cubicles, provide additional information on how they determined the fraction of leaking material that becomes airborne, and information describing if the contribution of other radionuclides besides Kr, Xe, Br, I, and tritium were considered in the airborne activity analysis and why only those radionuclides were selected. The staff also requested that the applicant provide additional radionuclides beyond only tritium for the fuel handling area (both during normal operation and refueling) and to provide additional clarifications.

In its initial response to **RAI 23-7929, Question 12.02-7 (ML15223B087)**, the applicant indicated that, for airborne sources outside containment during normal operation, all radionuclides besides Kr, Xe, Br, I, and tritium were considered to be negligible because other radionuclides are considered to be retained in leaked water. In addition, the applicant also provided the assumptions for estimating the airborne concentrations for Kr, Xe, Br, I, and tritium. The fraction of the noble gasses assumed to become airborne from leaked fluid is 100 percent (which is the maximum amount possible). Tritium airborne concentrations are based on the guidance in RG 1.183 "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Appendix A. Since this guidance is used to calculate flashing fractions and the amount of Iodine that would become airborne from piping leaks in the engineered safety feature (ESF) system, it is considered acceptable for the amount of leaked tritium that would become airborne because some of the leaked fluid would run into sumps and be collected prior to evaporation. However, the applicant assumed that 0.1 percent of halogens in cold liquid and 10 percent of halogens in hot liquid would become airborne based on NUREG-0409, "Iodine Behavior in a PWR Cooling System Following a Postulated Steam Generator Tube Rupture Accident" and European guidance all related to the behavior of Iodine in RCSs during SG tube rupture accidents. It is unclear to staff why it is appropriate to use this guidance to determine the amount of Iodine that becomes airborne during a normal piping leak.

The applicant indicated that the airborne activity concentrations in containment are based on the TS allowable undetected leakage rate, which is the maximum amount allowed for plant operation. Staff finds it acceptable to base the airborne activity concentration in containment on the technical specifications allowable undetected leakage rate limit, because it is the maximum amount of RCS leakage allowed and would result in higher airborne activity concentrations than would normally be expected. The applicant also provided additional information regarding how the airborne concentrations and minimum heat, ventilation, and air conditioning (HVAC) flow rates were calculated for other cubicles that could be anticipated to potentially contain significant quantities of airborne radioactive material. Finally, the applicant provided information on how airborne activity in the fuel handling area was calculated and indicated that only tritium was considered for the fuel handling area because tritium was considered as the most dominant radionuclide for airborne activity in fuel handling area. A review of other SFP airborne radioactivity calculations shows that tritium is a significantly larger contributor to the limited dose received from the inhalation of airborne activity in the SFP area and that other airborne radionuclides result in a negligible increase in the total derived air concentration limit fraction. Therefore, the staff finds it acceptable to only consider tritium.

In its revised response to **Question 12.02-7 (ML15316A472)**, the applicant stated it made a revision to correct an editorial error. However, neither the initial, nor the revised response provided the ventilation rate for the fuel handling area, adequate justifications for the use of SG tube rupture guidance for determining airborne halogen activity concentrations for pipe leaks, or adequate information regarding why it is appropriate to assume all radionuclides except for tritium and isotopes of Kr, Xe, Br, and I, do not become airborne outside containment. In addition, the applicant did not indicate how they determined the airborne activity of these other radionuclides inside containment. Also, the applicant did not include the assumptions for the amount of radioactivity of different nuclides that become airborne from leaks (the partition factors for different radionuclides) in the DCD. Therefore, the **staff closed Question 12.02-7 as an unresolved item** and issued **RAI 321-8353, Question 12.02-20 (ML15329A236),** requesting that the applicant provide this information. In addition, the staff requested that the applicant also provide additional information on compliance with 10 CFR 20.1701, "Use of process for engineering controls," and 10 CFR 20.1702, "Use of other controls," based on the initial response and to make additional clarifications and corrections.

In its response to **RAI 321-8353, Question 12.02-20, (ML16068A288)**, the applicant provided the ventilation flow rate for the fuel handling area in the auxiliary building and proposed adding this information into DCD Tier 2, Table 12.2-26 ([sheet] 2 of 8). Based on staff confirmatory calculations, the applicant's provided ventilation flow rate is sufficient to maintain airborne tritium concentrations to the levels specified in DCD Tier 2, Table 12.2-23, and is, therefore, acceptable.

The applicant indicated that the basis for partition factors of 0.1 percent for halogens (Iodine isotopes and Br-84) in cold fluid (less than 120 °F [48.9 °C]) was based on various guidance documents including WASH-1258, Volume 2, “Analytical Models and Calculations of the Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion ’As Low As Practicable’ for Radioactive Material in Light-Water Cooled Nuclear Power Reactor Effluents.” While the purpose of WASH-1258 is for effluent releases and not airborne activity concentrations in the plant, WASH-1258, Volume 2, does specify that for determining the amount of iodine that becomes airborne from leaked fluid to use a partition factor of 0.1 percent for the auxiliary building because the fluid would be expected to have a low iodine concentration. Iodine partition factors change significantly based on several factors, including temperature and concentration.

The staff reviewed several documents to determine if the use of a partition factor of 0.1 percent for iodine isotopes would be appropriate for leaked fluid outside containment (such as those portions of the CVCS system outside containment) with the iodine concentrations of the APR1400, with the assumed 0.25 percent failed fuel, including Electric Power Research Institute (EPRI)-NP-1271, “Nuclear Power Plant Related Iodine Partition Coefficients.” The staff determined that, for cold leakage outside containment, the iodine concentrations were low enough that the partition factor of 0.1 percent was an acceptable assumption and should bound different fluid conditions at the plant, as long as the fluid temperature is less than 120 °F (48.9 °C) as provided in the proposed DCD Tier 2, Subsection 12.2.2.4, “Basis and Assumptions for Partition Factors,” (added in the response) and the concentrations of the fluids outside containment based on 0.25 percent failed fuel are not exceeded. Therefore, the staff accepts the use of a 0.1 percent partition factor for iodine isotopes for determining the normal operation (0.25 percent failed fuel) airborne activity concentrations in cold fluid outside containment in the APR1400 design.

Iodine concentrations in the RCS are significantly higher than Br-84 and there are no plant processes that would significantly increase the concentration of Br-84 in plant fluid compared to the iodine radionuclides downstream of the RCS. In addition, many of the iodine isotopes have much lower intake limits and have been known to be a much more significant source of airborne exposure to plant workers than Br-84. Also, DCD Tier 2, Table 12.2-23 shows the estimated airborne radioactivity concentrations in containment during normal operation (based on 0.25 percent failed fuel) and Br-84 is insignificant compared to the sum of the derived air concentration limit fractions and has a DAC fraction value more than a million times less than I-131. Br-84 has a half-life of approximately 32 minutes, which is significantly shorter than most of the significant iodine isotopes and decays to a stable isotope, Kr-84. Based on the above, the staff concludes that Br-84 would be a negligible contributor to any dose received by workers associated with airborne radioactivity or the intake of airborne radioactive material by plant workers. Since, in reality, the Br-84 concentration will be insignificant compared to other radionuclides, the staff accepts the applicant’s use of a 0.1 percent partition factor for Br-84 in cold fluid outside containment. Since the leakage inside containment may contain untreated reactor coolant and will likely be at higher temperatures, the applicant calculates airborne concentrations for all radionuclides inside containment using the flashing factor approach described in RG 1.183, Appendix A, Section 5 (except for noble gases for which 100 percent of the activity will become airborne). This is an acceptable approach.

In its response to Question 12.02-20, the applicant also included other clarifications and DCD updates such as specifying the room numbers for the cubicles identified in Table 12.2-23 (cubicles likely to contain significant airborne radioactivity).

Finally, in its response to Question 12.02-20, the applicant indicated that, other than noble gases, halogens, and tritium, no other radionuclides are considered for airborne activity concentrations outside containment because most of the systems outside containment are secondary systems and contain little radioactivity. The staff does not agree with this position and operating experience shows that other radionuclides are found to become airborne in nuclear power plants.

In **Revision 1 of its response to Question 12.02-20 (ML16251A523)**, the applicant re-calculated the airborne radioactivity concentrations and DAC fractions considering a partition factor of 0.005 for other nuclides (primarily particulates) outside of containment (in the auxiliary and compound buildings). While the applicant indicated that the partition factor of 0.005 is based on NUREG-0017, the staff could not find this information in NUREG-0017. However, the fluids in systems outside containment are expected to be less than 212 °F (100 °C). As a result, flashing of the fluid to steam should not occur and the fraction of particulates that becomes airborne would be expected to be less than that of iodine in cold fluid. The partition factor of 0.005 is greater than the 0.001 partition factor used for iodine in cold fluid (discussed above); as a result, the staff views the partition factor of 0.005 for these other radionuclides to be a conservative assumption and is therefore, acceptable.

As a result of this change, the applicant also increased the ventilation flow rate for valve room 085-P16 in the compound building from 1,444 cubic meters per hour (850 cubic feet per minute) to 1530 cubic meters per hour (900 cubic feet per minute), in order to maintain the DAC fraction in this room to below a value of 0.1. No other minimum required ventilation flow rates were changed as a result of this response.

In its response to **RAI 207-8247, Question 12.02-18 (ML15343A410)**, the applicant also made corrections to DCD Tier 2, Table 12.2-26 to include corrected room volumes, leak rates, and HVAC flow rates. In addition, in its **response to RAI 390-8479, Question 12.02-27 (ML16152B034)**, the applicant made minor corrections to airborne radioactivity concentrations and calculated DAC fractions (as well as source term dimension corrections for the IRWST). The changes in these two RAI responses were made in order to make corrections to the original DCD information, and the staff found these changes acceptable. The changes will be included in a future DCD revision. **These changes will be tracked as confirmatory items.**

As a result of all of these changes, the DAC fraction for each of the main cubicles identified in DCD Tier 2, Tables 12.2-23 and 12.2-26 remains below, and in many cases well below, 1.0. While not all cubicles are identified in Table 12.2-23 and Table 12.2-26, many of the cubicles expected to contain the more significant quantities of airborne radioactive material are identified and the information shows that the ventilation system is adequate to maintain airborne activity levels below a DAC fraction of 1.0, with an assumed 0.25 percent failed fuel fraction. In addition, DCD Tier 2, Section 9.4.2, "Fuel Handling Area HVAC System," Section 9.4.5, "Engineered Safety Feature Ventilation System," and Section 9.4.7, "Compound Building HVAC System," specify that airflow is from areas of low potential radioactivity to areas of higher potential radioactivity and that the systems are designed to prevent the spreading of airborne radioactivity within the plant and to maintain the airborne radioactivity levels in all of these areas below the DAC values specified in 10 CFR Part 20, Appendix B. This information is also provided in DCD Tier 1, Section 2.8, “Radiation Protection,” which states that ventilation systems for the radiological controlled areas are designed to keep the radiation exposure below the limits specified in 10 CFR Part 20, Appendix B, and the acceptance criteria to ITAAC 2 in Table 2.8-2, "Radiation Protection ITAAC," states that the ventilation systems of radiologically controlled areas shall limit concentrations of airborne radionuclides to below the concentrations provided in 10 CFR Part 20, Appendix B (See the response to **RAI 116-8054, Question 14.03.08-11 (ML16034A204)**). The 10 CFR Part 20, Appendix B values referred to in Tier 1 are the calculation of the DAC fractions. Therefore, the ventilation system is designed to ensure that all areas of the plant, besides reactor containment during operation, is maintained below a 1.0 DAC fraction.

In the event that significant airborne radioactivity is present in an area due to unexpected operational conditions or if access to the inside of containment during operation is needed, the radiation protection and ALARA programs will ensure that adequate controls (including the use of respirators, if appropriate) are used to ensure worker dose is ALARA and that all regulatory requirements are met (see SER Sections 12.1 and 12.5 for a discussion of the radiation protection and ALARA programs). The applicant references Nuclear Energy Institute (NEI) 07-03A, "Generic FSAR Template Guidance for Radiation Protection Program Description," and NEI 07-08A, “Generic FSAR Template Guidance for Ensuring that Occupational Radiation Exposures Are As Low As (Is) Reasonably Achievable (ALARA),” which discuss the use of airborne monitors, respirators, and other criteria to limit the intake of airborne radioactive material. In addition, DCD Tier 2, Section 12.2.2.3 states that, except under abnormal conditions, general access areas normally have little, if any, airborne contaminants during normal operation. As a result of this information, and the response to Question 12.2-24, below, the staff determined that the airborne activity calculations, including DAC fractions, and minimum HVAC flow rates are acceptable. However, the proposed revisions to DCD Tier 2, Subsection 12.2.2.4.b, states that the partition factors provided apply to the auxiliary building, when according to the response those partition factors should apply to both the auxiliary building and compound building.

In **Revision 2 of its response to RAI 321-8353, Question 12.02-20 (ML16279A531)**, the applicant proposed updating DCD Tier 2, Subsection 12.2.2.4.b, to correct the editorial error and specify that the partition factors specified applied to both the auxiliary and compound buildings. The staff finds this response acceptable, and **Question 12.02-20 is being tracked as a confirmatory item**.

In addition, as a result of the Chapter 12 source term audit (as discussed in Section 12.2.4.1 of this SER), the staff asked additional questions related to the ventilation system flow rates and airborne radioactivity sources. One of the questions requested that the applicant specify if the ventilation system will be designed to the minimum flow rates used in the airborne activity calculations, specified in DCD Tier 2, Table 12.2-26, because the application was unclear if the ventilation system flow rates provided in Table 12.2-26 are the minimum design flow rates for each cubicle. In its response to these audit questions (ML15303A398), the applicant indicated that the minimum flow rates were necessary to maintain DAC fractions to less than 1.0, but the DCD did not clearly specify if the ventilation system will be designed to meet the minimum flow rates specified in Table 12.2-26. The staff issued **RAI 343-8420, Question 12.02-24 (ML15356A016)**, requesting that the applicant provide additional clarification on whether the ventilation system will be designed to the minimum values provided in DCD Tier 2, Table 12.2-

26. In addition, in Question 12.02-24, the staff requested that the applicant provide additional information on how the auxiliary building ventilation system source term was calculated.

In its initial response to **Question 12.02-24 (ML16050A023)**, the applicant specified that the HVAC flow rates provided in DCD Tier 2, Table 12.2-26 are the minimum required ventilation flow rates and that, for most areas, the ventilation flow rates are actually expected to be significantly higher than the values provided in Table 12.2-26. The applicant updated DCD Table 12.2-26 to specify that the flow rates provided are the minimum required ventilation flow rates. This is acceptable because the minimum HVAC flow rates provide sufficient ventilation flow to ensure that all areas of the plant remain below a DAC fraction of 1.0. The areas outside containment with airborne radioactive concentrations near one DAC that could potentially meet the definition of an airborne radioactivity area in 10 CFR 20.1003 (i.e., an area with a concentration that is less than one DAC but which an individual could have an intake exceeding 12 DAC-hours, in a week) are not areas that are expected to be frequently accessed. In addition, airflow is from areas of low airborne radioactivity to high airborne radioactivity, so that the spread of airborne radioactivity from these areas to less radioactive areas will be limited. This is acceptable because areas with a high potential to be airborne radioactivity areas are not areas expected to be accessed frequently or for long periods of time during normal operation (except if special maintenance or activities were needed, in which case airborne activity intake will be controlled by the radiation protection and ALARA programs).

Also, in its response to Question 12.02-24, the applicant explained how it determined the shielding design for the HVAC and charcoal filters. The applicant explained that they took the radionuclide concentrations from the PWR-GALE (gaseous and liquid effluent) code (summarized in DCD Tier 2, Table 11.3-1), “Expected Gaseous Radioactive Effluents During Normal Operations Including AOOs (Bq/yr),” assumed that the PWR-GALE code results were based on 0.12 percent fuel defect, adjusted the values by the ratio of 0.25 percent, percent and back calculated the PWR-GALE code results to determine the source term for the auxiliary building filters. However, the PWR-GALE code is not based on a particular failed fuel percentage, and the applicant provided no basis for assuming that the PWR-GALE code is based on 0.12 percent fuel defect. Therefore, the response was unacceptable and the applicant submitted **Revision 1 of its response to Question 12.02-24 (ML16124B177)**.

In Revision 1, the applicant recalculated the source term for the filters by taking the ratio of the 0.25 percent failed fuel source term in DCD Tier 2, Table 12.2-5 and the RCS source term from the GALE code in DCD Tier 2, Table 11.1-9 and then ratio the GALE code results, accordingly, to back calculate the source term for the auxiliary building filters. The I-131 source term went from 5.57E+06 Becquerel (Bq) (0.15 millicuries) to 1.20E+09 Bq (32.4 millicuries) as the result of this change. The staff reviewed the applicant’s calculations in the response and determined that they are an acceptable method for calculating the filter loading in the auxiliary building. While the revised response indicated that there was a significantly higher source term than what was provided in the initial response, the applicant did not provide any information regarding how the change impacted the radiation shielding and zoning design. In addition, it was unclear if similar changes needed to be made to calculate the filter loading of the compound building ventilation system filters.

In **Revision 2 of its response to Question 12.02-24 (ML16174A142)**, the applicant indicated that the corrected auxiliary building source term results in an increase in the radiation zoning and shielding for the auxiliary building air cleaning unit (ACU) filter room. The applicant proposed the appropriate revision to the DCD Tier 2, Chapter 12 radiation zone figures and Table 12.3-4, to address the increase shielding thickness requirements. The staff determined that the changes are acceptable because the shielding and zoning for the auxiliary building ACU filter room is adequately based on an acceptable filter source term.

For calculating the source term for the compound building filters, the applicant assumed that the compound building filters would have the same source term as that of the auxiliary building filters. Since the compound building filters would not be expected to have as high of a source term as that of the auxiliary building, the staff considers this assumption acceptable. The radiation zoning in the compound building increased from Zone 2 to Zone 6, as a result of the re-calculation. In the response to the question, the applicant also proposed providing the minimum required shielding thicknesses for the compound building exhaust air cleaning unit (ACU) room in DCD Tier 2, Table 12.3-4. The staff determined that these changes are acceptable.

Based on the above, the **Revision 2 response to Question 12.02-24, is being tracked as a confirmatory item**, pending the proposed DCD revisions.

12.2.4.3 Accident Source Terms

This section discusses the approach used in the APR1400 DCD for determining post-accident source terms, which are to be used to determine appropriate post-accident shielding and to ensure that the worker doses will not exceed dose limits during an accident as required by 10 CFR 50.34(f)(2)(vii) and as described in NUREG-0737 and SRP Section 12.2 and Section 12.3-12.4, “Radiation Protection Design Features.” It also discusses radiation sources and shielding barriers as they relate to main control room (MCR) dose and compliance with GDC 19.

The shielding and zoning for the post-accident sources discussed in this section are used to determine vital area access shielding and doses for workers during accident conditions, as discussed in Section 12.3 of this SER. The complete review of compliance with GDC 19, including contributions from airborne radioactivity inside the MCR due to an accident, is discussed in Chapter 15 of this SER. The post-accident source terms are also used to calculate the accident dose rates for equipment qualification, however, for the purposes of equipment qualification a loss of coolant accident (LOCA) is not the limiting accident for all equipment. (Equipment qualification information is discussed in Section 3.11 of this SER.)

The approach used for calculating post-accident source terms is in DCD Tier 2, Section 12.2.3, “Sources Used in NUREG-0737 Post-Accident Shielding,” and Chapter 15. The core activity release model for a design-basis accident (DBA) is based on the source term model from RG 1.183. The applicant assumed that, following a design-basis LOCA, core releases occur to containment based on the release fractions for gap releases and early in-vessel releases, specified in Table 2, “PWR Core Inventory Fraction Released Into Containment,” of RG 1.183 and DCD Tier 2, Table 12.2-24 “Source Terms for Post-Accident Shielding Analysis.” These release assumptions are used to calculate the activity levels inside containment during an accident, the activity levels in systems that circulate post-accident fluid (the safety injection, containment spray, and shutdown cooling systems), and release rates to the atmosphere.

However, while the application provides the general methodology for calculating post-accident source terms, it does not provide source term data in isotopic concentrations or energy levels for any of the post-accident sources, contrary to SRP Section 12.2. The staff issued **RAI 207-8247, Question 12.02-16 (ML15295A505)**, requesting that the applicant clarify the source term information for systems that recirculate post-accident fluids outside containment and for filters in the control room and technical support center (TSC) emergency ventilation system, as well as assumptions and parameters that were used to develop the sources. In RAI 207-8247, Question 12.02-16, the staff also requests the applicant to indicate if there are any ventilation filters (for normal operation or accident conditions) located within the MCR or TSC area, provide shielding thicknesses for the MCR and TSC, and to ensure that information in the DCD is consistent.

In its response to **Question 12.02-16 (ML15343A410)**, the applicant provided the source terms for post-accident sources in DCD Tier2, Table 12.2-24. The source terms include the post-accident recirculating water, airborne source term in containment, and the MCR emergency makeup ACU filter inventories.

The source term for the post-accident recirculating water assumes that all radionuclides except noble gases are released to the containment atmosphere until the early in-vessel release phase of the alternative source term (described in Chapter 15) are mixed instantaneously and homogeneously in the primary containment sump water at the start of the accident. The staff verified the post-accident recirculating water source term and determined that it is acceptable. The source term for each component is based on the volume of each component (including piping). The recirculating fluid is found in the shutdown cooling system, safety injection system, and containment spray system. Other plant systems, such as the CVCS system are not required to operate during DBAs and therefore are not expected to have accident source terms.

The source term for the control room emergency filters is based on a design basis LOCA, and NUREG/CR-6604, “RADTRAD: A Simplified Model for RADionuclide Transport and Removal and Dose Estimation," December 1997. The Radionuclide Transport, Removal, and Dose (RADTRAD) computer program was used to calculate the filter source term (as well as the airborne activity inside the MCR, as described in Chapter 15). The MCR filters are located on the floor above the MCR and TSC. There is 1.5-feet (about 0.46 meters) of concrete shielding provided to shield the operators in the MCR and TSC from the filters, as specified in DCD Tier 2, Table 6.4-1, “The Accident Radiation Source Description and Radiation Shielding Design for MCR and TSC.” There are no other filters associated with filtering the post-accident MCR and TSC atmosphere. There are no other direct radiation streaming paths to the MCR or TSC of any significance, because there are no major containment penetrations at the elevation and direction of the MCR or TSC and there is a large distance and multiple shields between the MCR and TSC and any significant radiation sources, including those inside containment. The staff confirmatory shielding calculations for the dose rate inside the MCR from the filters was about 800 mrem (8 mSv) for the duration of the accident (assumed to be 30 days, as specified in NUREG-0737). This is less than the 1,020 mrem (10.2 mSv), specified by the applicant in DCD Tier 2, Table 15.6.5-14 “Radiological Consequences of a Large Break LOCA.” However, in **Revision 1 of its response to RAI 207-8247, Question 12.02-16 (ML16230A440)**, the applicant revised the source term for the MCR filters because the previous source term was based on incorrect atmospheric dispersion assumptions. In reviewing the revised response, the staff noted apparent errors in the source term. Most notably, the one-week source term appeared to contain erroneous information, which was inconsistent with the other information in the table. The applicant is expected to provide another revision of the response in order to resolve this issue. **Question 12.02-16, is being tracked as an Open Item** until the proposed DCD corrections have been made.

In addition, in its revised response to Question 12.02-16, the applicant did not provide any information indicating if the design prevents direct or near direct radiation streaming from the emergency filters into the control room envelope through the openings of the ventilation ducting or other penetrations into the control room envelope. The staff **issued RAI 390-8479, Question**

12.02-26 (ML16032A388), requesting that the applicant address this issue and provide the density of concrete assumed in the calculations for the dose to operators in the control room envelope and for all other shielding calculations in the plant.

In its response to **Question 12.02-26 (ML16142A021)**, the applicant specified that in the shielding analysis performed for the APR1400, concrete is assumed to have a density of 2.242 grams per cubic centimeter (140 pounds per cubic feet) and proposed adding this information to DCD Tier 2, Section 12.3.2.2, “Shielding Analysis.” The staff finds that this is a reasonable value for concrete shielding density, and is, therefore, acceptable.

The applicant also indicated that the location of the emergency air cleaning units are located above the computer room and the other is located above the TSC. The HVAC openings for the MCR are directly above the MCR. Therefore, there is no direct streaming path into the MCR. In **Revision 1 of its response to Question 12.02-26 (ML16176A372)**, the applicant added a paragraph to DCD Tier 2, Table 6.4-1 to provide information specifying that there is no direct or near direct streaming through the HVAC penetrations. The applicant also proposed to make several other corrections in the response and proposed adding information to DCD Tier 2, Subsection 12.3.2.3, “Shielding Design,” specifying that two layers of concrete shield plugs are installed around the reactor vessel. The information indicates that the shield plugs are used to reduce dose rates on the operating floor. Adequate air cooling is provided to the area between the reactor and the shield plugs and the primary biological shield to ensure the shielding capability and structural integrity are not degraded from heat. These shield plugs are an acceptable design feature to reduce neutron and gamma ray streaming from the reactor core to the operating floor and are therefore acceptable. Therefore, the staff finds this response acceptable and **Revision 1 of the response to RAI 390-8479, Question 12.02-26, is being tracked as a confirmatory item**, pending the incorporation of the proposed DCD revisions.

In its response to **RAI 368-8470, Question 14.03.08-14 (ML15343A410)**, Part 6, the applicant indicated that, upon detection of high radiation in the outside air intakes, the outside air intake isolation dampers in the outside air intake having the higher radiation level close automatically and will re-open after a pre-determined time period. It is unclear to the staff if the occasional re-opening of the MCR dampers was appropriately considered in the Chapter 15 and Section 6.4 MCR dose analysis and in the MCR filter loading source term that the applicant provided in the response to RAI 207-8247, Question 12.02-16 (see above), for the proposed changes in DCD Tier 2, Table 12.2-24, sheets 13 and 14. The applicant is expected to provide more information upon resolution of Question 14.03.08-14, and correction of the MCR filter source term in the response to Question 12.02-16. **RAI 368-8470, Question 14.03.08-14 is being tracked as an Open Item in Section 14.3.8**, “ITAAC for Radiation Protection,” of this SER.

Based on the above review, except for the MCR filters, the staff determined that the post-accident source terms provided in the proposed update to DCD Tier 2, Table 12.2-24 are acceptable. See Section 12.3 of this SER for the evaluation of the dose to workers in accessing vital areas, as a result of the accident sources. As discussed above, the review of the MCR filter source term is incomplete, pending the resolution of the open RAI questions, discussed above.

Other aspects of the post-DBA review, including the review of the dose to MCR operators from the airborne radioactivity inside the MCR and doses at the site exclusion area boundary are discussed in Chapter 15 of this SER. Based on the resolution of the above issues and issues discussed in Chapter 15, the staff will make a determination regarding MCR dose and compliance with GDC 19 in Chapter 15 of this SER.

12.2.5 Combined License Information Items

The following is a list of items from Table 1.8-2 of the DCD:

COL 12.2(1): The COL applicant is to provide any additional contained radiation sources, such as instrument calibration radiation sources, that are not identified in Subsection 12.2.1.

12.2.6 Conclusion

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, or justified appropriate alternative methodologies. For post-accident shielding for vital area access, the applicant used the source terms in NUREG-0737 and RG 1.183. The staff reviewed and verified many of the sources in the APR1400 design and determined that they are acceptable.

During power operation, the greatest potential for personnel dose is inside the containment from N-16, noble gases, and neutrons. Outside the containment and after shutdown inside the containment, the primary sources of personnel exposure are fission products from fuel clad defects and activation products, including activated corrosion products.

The main sources of airborne radioactivity are from sources located inside containment, the auxiliary building (including the SFP area), and the compound building. Leakage from the equipment compartments constitutes the main airborne sources in the plant. Leakage from stored spent fuel assemblies and evaporative losses from the SFP are the main sources of airborne activity in the SFP area. In these areas, the applicant has provided a tabulation of the maximum expected routine radioactive airborne concentrations for normally occupied areas.

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 10 CFR 52.47(a)(5), as it relates to the information on radiation sources provided by the applicant. Upon the resolution of the **Open Items tracked under Questions 12.02-2, 12.02-16, 12.02-22, 12.02-23, and 12.02-25,** the staff will determine if the APR1400 design meets the requirements of 10 CFR Part 20, 10 CFR 50.34(f)(2)(vii), 10 CFR 52.47(a)(5), GDC 19 and GDC 61, "Fuel Storage and Handling and Radioactivity Control." The staff also determined that it is 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 Tier 2, Section 12.3, "Radiation Protection Design Features," and Section 12.4, "Dose Assessment and Minimization of Contamination," because NUREG-0800, Section 12.3-12.4 combines both sections.

12.3.1 Introduction

This section focuses on radiation protection design features, including the equipment used for assuring that occupational radiation exposures (ORE) will be as low as is reasonably achievable (ALARA). Dose rates during normal operation, AOOs, 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, 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., SG tube plugging), refueling, and in-service inspection provide a measure of the effectiveness of the proposed design features, in reducing overall area dose rates.

12.3.2 Summary of Application

DCD Tier 1: The DCD Tier 1 information associated with this section includes Tier 1 Section 2.4.7, “Leakage Detection System,” Section 2.7.3, “HVAC Systems,” Section 2.7.6.4, “Process and Effluent Radiation Monitoring and Sampling System,” Section 2.7.6.5, “Area Radiation Monitoring System,” and Section 2.8, “Radiation Protection,” and consists of design features which demonstrate compliance with the occupational radiation safety requirements of 10 CFR Part 20, including those Tier 1 sections, which address radiation shielding and zoning for radiological areas of the plant, for radiation monitors including the containment high radiation accident monitors, the MCR ventilation accident radiation monitors, and fuel handling area radiation monitors. In addition, these sections identify relevant key design features, seismic classifications, interlocks, Class 1E power supplies, and ITAAC. The staff’s evaluation of the ITAAC and additional evaluation of the Tier 1 information is provided in Section 14.3.8 of this SER.

DCD Tier 2: The applicant has provided a DCD Tier 2 system description in Section 12.3-12.4, 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 APR1400 application is a projected annual personnel dose assessment for the APR1400.
>
> Health physics facilities are provided as part of the plant design, including health physics work areas, decontamination facilities, and storage areas.
>
> The inner compartment of the containment building contains the SGs, reactor coolant pumps, and primary loop piping. The containment building outer compartment houses support equipment. Shielded compartments are provided for CVCS components located outside of the secondary shield in containment, such as the letdown heat exchanger.
>
> A hot machine shop is provided in the compound building so that maintenance can be performed on radioactive and contaminated equipment. The hot machine shop allows for maintenance and repair activities to be performed in a lower-radiation area.
>
> 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, to minimize releases of radioactivity to the environment.

Potential very high radiation areas (VHRAs) (areas with a dose rate of 500 rad (5 Gray [Gy]) or more per hour at 1 meter [3.3 feet] from a radiation source or any surface through which the radiation penetrates) in the containment building during normal and refueling operations include the reactor cavity, in-core instrument (ICI) cavity, and core debris chamber area. Potential VHRAs in the auxiliary building include the purification ion exchangers, purification filters, and volume control tank room. Potential VHRAs in the compound building include the spent resin long-term storage tank room, the charcoal delay bed room, and the instrument calibrator facility, during calibration activities when the calibration source is unshielded. In addition, the refueling canal, cask load pit, SFP, and fuel transfer tube are VHRAs when fuel is present. The applicant provided a complete list of areas potentially greater than 100 rad per hour (1 Gy/hr) in DCD Tier 2, Table 12.3-5, “Areas Potentially Greater than 1 Gy/hr,” including potential VHRAs.

Radiation zones for each plant area are defined by the dose rate in the area considering sources within each area as well as contributing dose rates from sources in adjacent areas. Radiation zone categories are described in DCD Tier 2, Table 12.3-2, "Normal Operation Radiation Zone Designations." The radiation zone figures provided in DCD Tier 2, Section 12.3 provide the dose rates throughout the plant at 30 centimeters (1 foot) from the radiation source or any surface that the radiation penetrates.

The area radiation monitoring instrumentation for use during normal operation and AOOs 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 area radiation monitoring instrumentation also warns of possible equipment malfunctions, operator errors, or other radiological abnormalities in specific areas and furnishes information to supplement radiation surveys. Safety-related area radiation monitoring equipment is seismically qualified, environmentally qualified, and powered by a Class 1E power supply. The area radiation monitors are listed in DCD Tier 2, Table 12.3-6 “Area Radiation Monitors.” Process and effluent radiation monitors are described in DCD Tier 2, Section 11.5, “Process and Effluent Radiation Monitoring and Sampling Systems.” The DCD Tier 2, Section 11.5, Figures 11.5-2A through 11.5-2Z, show the location of all plant radiation monitors (including area monitors), as well as the location of instrument electronics/displays and alarms. On detection of high radiation levels, all area radiation monitors will alarm in the MCR.

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 the dose from airborne radioactivity will generally not be a significant contribution to the total dose. The applicant estimates a total annual ORE for an APR1400 unit of 0.585 person-Sv (58.5 person-rem). This includes the activities of reactor operations and surveillance, routine maintenance, in-service inspection, special maintenance (such as SG re-tubing), waste processing, and refueling.

ITAAC: The ITAAC associated with DCD Tier 2, Section 12.3 and Section 12.4 are given in DCD Tier 1, Sections 2.4.7, 2.7.3, 2.7.6.4, 2.7.6.5, and 2.8.

TS: TS for the control of high radiation areas are addressed in DCD Tier 2, Chapter 16, "Technical Specifications," Section 5.7, “High Radiation Area.” TS for post-accident monitoring instrumentation are addressed in DCD Tier 2, Chapter 16, Section 3.3.11, “Accident Monitoring Instrumentation (AMI).”

12.3.3 Regulatory Basis

The relevant requirements of the Commission's regulations for this area of review, and the associated acceptance criteria, are given in Section 12.3-12.4 of NUREG-0800, and are summarized below. Review interfaces with other SRP sections can be found in Section 12.3-12.4 of NUREG-0800.

1. Section 20.1101(b) of 10 CFR 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. Section 20.1201 of 10 CFR, as it relates to occupational dose limits for adults.

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

4. Section 20.1301 of 10 CFR 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 AOOs.

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

6. Section 20.1601, "Control of access to high radiation areas"; of 10 CFR, 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. Section 20.1801, "Security of stored material," of 10 CFR, as it relates to securing licensed materials against unauthorized removal from the place of storage.

8. 10 CFR 50.34(f)(2)(vii), which requires the performance of radiation shielding design reviews to ensure the design permits adequate access to important areas and provides for protection of safety equipment from radiation, following an accident.

9. Part 50.34(f)(2)(xvii), of 10 CFR, as it relates to the requirement to provide instrumentation to measure, record and readout in the MCR containment radiation intensity (high level).

10. Part 50 of 10 CFR, 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 mSv (5 rem) to the whole body or the equivalent to any part of the whole body for the duration of the accident in accordance with 10 CFR 50.34(f)(vii).

11. Part 50 of 10 CFR, Appendix A, GDC 61, as it relates to occupational radiation protection aspects of fuel storage, handling, radioactive waste, and other systems that may contain radioactivity, designed to ensure adequate safety during normal and postulated accident conditions, with suitable shielding and appropriate containment and filtering systems.

12. Part 50 of 10 CFR, Appendix A, GDC 63, "Monitoring fuel and waste storage," as it relates to detecting excessive radiation levels in the facility.

13. 10 CFR Part 50, Appendix E “Emergency Planning and Preparedness for Production and Utilization Facilities,” Section VI.2(a), which requires radiation monitoring systems for reactor coolant radioactivity, containment radiation level, condenser air removal radiation level and process radiation monitor levels.

14. Section 50.68 of 10 CFR, "Criticality accident requirements," as it relates to procedures and criteria for radiation monitoring in areas where special nuclear material is stored and handled.

15. Section 52.47(a)(5) of 10 CFR, which requires that the DC application contain a description of the kinds and quantities of radioactive materials expected to be produced in the operation and the means for controlling and limiting radioactive effluents and radiation exposures within the limits set forth in 10 CFR Part 20.

16. 10 CFR 52.47(a)(22) as it relates to ensuring that information necessary to demonstrate how operating experience insights have been incorporated into the plant design.

17. Section 52.47(b)(1) of 10 CFR, which requires that a DC application contain the proposed (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the DC is built and will operate in accordance with the DC, the COL, the provisions of the Atomic Energy Act of 1954, as amended, and the NRC's regulations.

The following RGs, NUREGs, and industry standards provide information, recommendations, and guidance and in general describe a basis acceptable to the staff for implementing the requirements of the regulations identified above:

1. RG 1.7, Revision 3, “Control of Combustible Gas Concentrations in Containment,” as it relates to methods for determining gaseous radionuclides in containment following an accident.

2. RG 1.52, Revision 3, "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 ESF atmosphere cleanup systems operable under postulated DBA conditions, to be designated as "primary systems."

3. RG 1.69, Revision 0, "Concrete Radiation Shields and Generic Shield Testing for Nuclear Power Plants," as it relates to the requirements and recommended practices acceptable for construction of facilities that apply to occupational radiation protection shielding structures for nuclear power plants.

4. RG 1.97, Revision 4, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," as it relates to a method acceptable to the staff for complying with the Commission's regulations to provide instrumentation for radiation monitoring following an accident in a light-water-cooled nuclear power plant.

5. RG 1.183, Revision 0, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," as it relates to the assumptions and methods for evaluating doses to individuals accessing the facility during and following an accident in accordance with NUREG-0737, Item II.B.2.

6. RG 4.21, Revision 0, "Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning," as it describes methods acceptable to the staff for complying with the requirements of 10 CFR 20.1406.

7. RG 8.2, Revision 2, "Guide for Administrative Practices in Radiation Monitoring," as it relates to general information on radiation monitoring programs for administrative personnel.

8. RG 8.8, Revision 3, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable," as it relates to actions taken during facility design, engineering, construction, operation, and decommissioning to maintain ORE ALARA in accordance with 10 CFR 20.1101(b) and the definition of ALARA in 10 CFR 20.1003, concerning the radiation protection information to be supplied in DCD Tier 2, Chapter 12.

9. RG 8.10, Revision 1-R "Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as Is Reasonably Achievable," as it relates to the commitment by management and vigilance by the radiation protection manager and staff to maintain ORE ALARA in accordance with 10 CFR 20.1101(b) and the definition of ALARA in 10 CFR 20.1003.

10. Regulatory Guide 8.19, "Occupational Dose Assessment in Light-Water Reactor Power Plants Design Stage Man-Rem Estimates." as it relates to a method acceptable to the staff for performing an assessment of collective occupational radiation dose as part of the ongoing design review process so that such exposures will be ALARA.

11. Regulatory Guide 8.25, “Air Sampling in the Workplace,” as it relates to a method acceptable to the staff for continuous monitoring for airborne radioactive materials in plant spaces.

12. RG 8.38, Revision 1, “Control of Access to High and Very High Radiation Areas of Nuclear Plants,” as it relates to the physical controls for personnel access to high and very high radiation areas.

13. ANSI/ANS-HPSSC-6.8.1-1981, “Location and Design Criteria for Area Radiation Monitoring Systems for Light Water Nuclear Reactors,” as it relates to criteria for the establishment of locations for fixed continuous area gamma radiation monitors and for design features and ranges of measurement.

14. ANSI N13.1-1999, “Sampling and Monitoring Releases of Airborne Radioactive Substances from the Stacks and Ducts of Nuclear Facilities,” as it relates to the principles that apply in obtaining valid samples of airborne radioactive materials, and acceptable methods and materials for gas and particle sampling.

15. ANSI/ANS-6.4-1997 (R2004), “Nuclear Analysis and Design of Concrete Radiation Shielding for Nuclear Power Plants,” as it relates to requirements and recommended practices for the construction of concrete radiation shielding structures.

12.3.4 Technical Evaluation

The staff reviewed the radiation protection design features, dose assessment, and minimization of contamination design considerations in DCD Tier 2, Sections 12.3 and 12.4, for consistency with the guidance in RG 1.206 and the criteria in SRP Section 12.3-12.4. The purpose of this review was to ensure that the applicant had either committed to follow the guidance of the RGs and applicable staff positions, or offered acceptable alternatives. Where the DCD is consistent with the guidance in 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 features to help maintain OREs ALARA in accordance with the guidance in RG 8.8 and the requirements of 10 CFR 20.1101(b). These include facility design, shielding, ventilation, and area and airborne radiation monitors. These design features are founded on the ALARA design considerations described in DCD Tier 2, Section 12.1 and discussed in Section 12.1 of this SER.

12.3.4.1.1 Facility Design Features

The main sources of radiation inside containment are the reactor vessel, the primary loop components and associated piping. The primary shield, in conjunction with the secondary shield, reduces radiation levels from the reactor components and primary loop components and piping. Removable insulation is used around the reactor coolant piping to reduce radiation exposure when necessary. The removable insulation is held in place mainly by quick-application type buckle fasteners to permit for easy inspection of reactor vessel nozzle welds.

Platforms are provided around the reactor coolant pumps to facilitate the replacement of reactor coolant pump seals. The seals are a cartridge type reactor coolant pump seal that can be easily removed and repaired in a low radiation area and easily replaced.

The SG design incorporates features to facilitate maintenance and inspection that reduce ORE, include sizing the manway openings to facilitate access and locating manways in areas that allow for use of remote equipment for inspection and maintenance and handholes which facilitate access and handling of SG parts.

Thermally treated alloy 690 base metal is used for reactor coolant pressure boundary applications, including SG tubing material. Thermally treated alloy 690 metal is used to reduce the possibility of intergranular stress corrosion cracking, which could reduce equipment failure and therefore reduce worker dose due to maintenance activities.

While DCD Tier 2, Subsections 12.3.1.3, "Source Term Control," and 12.4.1.1.1, "ALARA Design Features for Occupational Dose Reduction," note the use of low cobalt material and provision of features to prevent buildup of radioactive material, the cobalt contents discussed in the DCD and provided for some components in DCD Tier 2, Table 12.3-1, "Cobalt Contents of Finished Surface for Primary System Materials," appeared inconsistent with industry guidance for recommended cobalt levels. In addition, the DCD did not appear to provide the cobalt content for all relevant systems in contact with primary coolant. The staff issued **RAI 94-7999, Question 12.03-7 (ML15202A664)**, requesting that the applicant provide additional information on this topic and to justify why the specified cobalt content levels are acceptable, or modify them, as appropriate, to meet industry guidance.

In its response to **Question 12.03-7 (ML15252A446)**, the applicant proposed revising DCD Tier 2, Table 12.3-1. The revised table specified a cobalt content weight percentage for the reactor coolant pump of 0.05 maximum weight percent instead of 0.1 weight percent. In addition, the applicant specified the maximum cobalt weight percent of 0.05 for the surge line and control element drive mechanisms including the reactor vessel control element drive mechanism nozzles and a mean cobalt weight percentage of 0.015 for the SG tubes. The staff determined that these changes are consistent with industry guidance and the staff's guidance for limiting cobalt, including RG 8.8 and EPRI TR-016780, "Utility Requirements Document" (URD). However, the response did not specify any cobalt limitations for the reactor coolant pump impeller or for the metals in fuel assemblies. The applicant also indicated that small quantities of cobalt-base alloy will be used in bar, casting, or hard-facing of the control element drive mechanism, reactor vessel internal, pumps, or valves, but didn't provide any information regarding why cobalt based alloys were being used in these areas. Finally, the applicant indicated that the cobalt content of other components in contact with the RCS, other than those identified in DCD Tier 2, Table 12.3-1, would not be limited, but did not provide any basis or justification for why the cobalt content of these components need not be limited. In order to resolve these issues and to ensure that the cobalt content and therefore, the presence of activated cobalt is minimized to the extent practicable, the **staff closed Question 12.03-7, as an unresolved item** and issued **RAI 309-8389, Question 12.03-48 (ML15320A157)**, requesting that the applicant provide this additional information.

In its response to **RAI 309-8389, Question 12.03-48 (ML16148B021)**, the applicant proposed updating DCD Tier 2, Table 12.3-1 to specify a cobalt content for the reactor coolant pump impeller of 0.05 weight percent (maximum). In addition, DCD Tier 2, Section 4.2.1.1, "Cladding," specifies that the fuel cladding is made of ZIRLO, which is known to have a very low cobalt content (the response specified a cobalt content of 0.002 weight percentage maximum for the fuel rods). The applicant also specified that cobalt-base hard-facing alloys are used to provide wear resistance only when no proven alternative exists. These areas where cobalt-base hard-facing alloys are used are the pin in the motor assembly of the control element drive mechanism, the latch and link in the motor assembly of the control element drive mechanism, the core support barrel snubber lug inner surface and fuel alignment plate key inner surface in the reactor vessel internals, and valve hard-facing. These components take up a very small surface area and many of them are not in areas subject to a high neutron flux. Therefore, the staff determined that the cobalt content of primary system materials provided in the proposed revision to DCD Tier 2, Table 12.3-1 and added Table 12.3-2, provide maximum cobalt content values consistent with EPRI TR-016780 (except for a few components, with a small surface area that would not be expected to result in large amounts of activated cobalt in the RCS) and appropriately limit cobalt to the extent practicable, in accordance with RG 8.8. As a result, the response to **Question 12.03-48, is being tracked as a confirmatory item** pending the incorporation of the proposed DCD updates.

DCD Tier 2, Subsection 5.2.3.2.1, "Reactor Coolant Chemistry," indicates that the COL applicant is to specify the version of EPRI's "Primary Water Chemistry Guidelines" that are to be used. However, the COL information item does not specify if the guidance used will ensure that radiation fields in the plant will be limited to the extent practicable. In addition, it is unclear if the APR1400 design includes provisions for connecting a zinc injection system, if needed. The staff issued **RAI 235-8275, Question 12.03-29 (ML15296A006),** requesting that the applicant provide this information.

In its response to **RAI 235-8275, Question 12.03-29 (ML15334A454)**, the applicant indicated that the APR1400 does not have a zinc injection system but has water chemistry pH control and ion exchangers in the CVCS in order to reduce dose rates. However, the applicant proposed COL Information Item 12.3(3) indicating that the COL applicant is to establish how the water chemistry pH control reduces radiation fields. There is no requirement for the design to contain a zinc injection system. Therefore, it is acceptable to rely on pH control and the CVCS ion exchangers to maintain the primary water source term and associated doses ALARA. It is acceptable for the COL applicant to provide information on how pH control will be used to reduce radiation fields. As a result, the staff determined that the response is acceptable, pending the proposed DCD changes. Therefore, RAI 235-8275, **Question 12.03-29 is being tracked as a confirmatory item**, until the proposed changes have been incorporated into the DCD.

In addition to the potential for activation products from the corrosion of components in contact with RCS fluid, plants with high fluid temperatures and high surface heat flux at the fuel clad have a portion of the total heat transfer to the coolant occur by sub-cooled nucleate boiling. This can lead to more severe duty on the fuel and surface boiling which is known to enhance the formation of corrosion product deposits (known as crud) on the cladding surface. The staff's calculations using the peak fuel assembly power indicated that the APR1400 is a high-duty core, subject to this type of corrosion product deposits on the cladding surface. Therefore, the staff issued **RAI 225-8254, Question 12.03-17 (ML15268A002)**, requesting that the applicant provide information on the design features to reduce crud buildup in the core, reduce dose rates, or provide methods to limit worker dose from these corrosion product deposits.

In its response to **Question 12.03-17 (ML16029A042)**, the applicant provided a calculation to determine if the APR1400 was a high-duty core, as described above. The calculation showed that the APR1400 was not a high-duty core. However, in performing the calculation, the applicant used a peaking factor of 1.2353 for the maximum assembly in the core. This value was significantly lower than the peaking factor value of 1.55 specified in DCD Tier 2, Section 12.3.2.3, for determining the maximum fuel assembly source term for radiation shielding purposes. This difference in peaking factor is the difference between the core being a high-duty core and medium-duty core.

In **Revision 1 of its response to Question 12.03-17 (ML16152B031)**, the applicant proposed updating DCD Tier 2, Subsection 12.3.2.3, to specify that the peaking factor of 1.55 is the ratio of the average power per unit length, while 1.2353 is the maximum assembly power during the first cycle of steady-state. For the APR1400, the maximum assembly in first cycle of steady-state would be the maximum assembly during the entire lifecycle of the plant, as described in APR1400-F-A-NR-13002-P, "The Effect of Thermal Conductivity Degradation on APR1400 Design and Safety Analysis." Therefore, it is acceptable to use the value of 1.2353 to determine if the APR1400 core is a high-duty core. In addition, it is conservative to use the value of 1.55 as the peaking factor for determining the radiation shielding and dose rates from the maximum fuel assembly. As a result, the staff accepts the applicant's calculation showing that the APR1400 is not a high-duty core and therefore, there is no need to describe design features to reduce dose rates from the potential buildup of corrosion products on the fuel. Therefore, **Question 12.03-17 is being tracked as a confirmatory item**, pending the incorporation of the proposed DCD revisions.

The APR1400 design includes other design features to limit radiation exposure, as discussed in DCD Tier 2, Section 12.3.1, "Facility Design Features." These design features include general arrangement design features, such as separating non-radioactive equipment from radioactive equipment, providing adequate space to work on equipment, and providing equipment staging areas. The design also includes personnel decontamination and changing areas and labyrinth access ways to high radiation areas. Equipment-specific design features include pump seals that are easily serviceable and removable to allow for repair work to be performed in low radiation areas; provisions to remove resins and filters remotely; piping running through shielded pipe areas, with the length of highly radioactive piping is limited to the extent practicable; and piping welding with butt welds to produce smoother surfaces in order to reduce the potential for crud buildup. DCD Tier 2, Section 12.3.1, provides a much more extensive list of design features to ensure that occupational exposure and exposure to members of the public are maintained ALARA.

While the application discusses design features for pumps and valves to ensure that doses to workers will remain ALARA, the information appeared incomplete and outdated. For example, the DCD indicated that lantern rings would be used to prevent leakage of valve packing. However, industry guidance indicates that lantern rings are not useful at preventing packing leaks and that fully engineered packing sets, valve stem material and finish specifications, and the use of live load packing on critical valves, and valves subject to thermal tapering, are methods that improve valve reliability and reduce leakage. Therefore, the staff issued **RAI 235-8275, Question 12.03-44 (ML15296A006)**, requesting the applicant to provide additional information and clarification regarding design provisions for pumps and valves to maintain occupational exposure ALARA and minimize contamination in accordance with 10 CFR 20.1406, including if electro-polishing will be used on major components in the RCS and elsewhere in the design. Electro-polishing can significantly reduce the dose rates from components, because it helps reduce the buildup of crud on component surfaces. In addition, DCD Tier 2, Subsection 5.2.5.1.2.3, "Valves," indicates that valves inside the containment that isolate the RCS from connecting systems during normal operation are equipped with stem leakoffs at the stem packing. DCD Tier 2, Figure 1.7-1, "Flow Diagram Symbols and Legend," shows the symbol for valves that use the stem leakoff line. The staff reviewed flow diagrams depicting the RCS, CVCS, shutdown cooling system, and other systems for key valves that may include stem leakoff lines. However, the staff was not able to locate any in the DCD figures. Therefore, the staff issued **RAI 235-8275, Question 12.03-45 (ML15296A006),** requesting that the applicant provide additional information regarding which valves use leakoff lines so that the leakage can be detected.

In its response to **Question 12.03-44 (ML15334A454)**, the applicant provided additional information on the design and use of valves and pumps. Information included that valves subject to frequent movement will be double packed with live loading packing, to ensure the lowest achievable leak rate possible and that metallic bellows and diaphragms are limited to valves with low-stroke length applications or infrequent movement. In addition, the applicant indicated that the reactor coolant pump casings are fabricated with austenitic stainless steel cladding and the finished surface of the cladding is machined to have a smooth surface roughness and to avoid radioactive contamination on the surface. Finally, the applicant indicated that electro-polishing would not be used on safety-related auxiliary pumps.

However, the applicant did not update the DCD with any of the information provided in its response. In addition, in its response, the applicant indicated that industry guidance recommends the use of leak off lines with graphite lantern rings. In its response to **RAI 235-8275, Question 12.03-45 (ML15334A454)**, the applicant proposed updating DCD Tier 2, Figure 5.1.2-3, "Pressurizer and POSRV Flow Diagram"; Figure 6.3.2-1, "Safety Injection/Shutdown Cooling System Flow Diagram"; and Figure 9.3.4-1, "Chemical and Volume Control System Flow Diagram," to show leak off lines on many of the valves in these systems. However, more recent industry guidance, such as EPRI's URD, Volume 2, Chapter 1, Section 12.2, "Valves," Subsection 12.2.2.10 and EPRI-TR-1000923, Section 4.2.5, recommend against the use of lantern rings except in limited circumstances, where leak monitoring is required, because this design could actually result in more leakage from plant components.

As a result, **the staff closed RAI 235-8275, Questions 12.03-44 and 12.03-45, as unresolved items** and issued **RAI 396-8463, Question 12.03-52 (ML16034A054)**, requesting the applicant to address the above issues. In addition, in RAI 396-8463, Question 12.03-52, the staff requested that the applicant provide additional information on the cladding of pumps, other than the reactor coolant pumps, to specify if they are designed to reduce material deposition on the pump surface and if electro-polishing would be used on any components in the plant such as RCS piping, SG channel heads and divider plates, or the reactor cavity and transfer canal liners. Finally, in RAI 396-8463, Question 12.03-52, the staff requested that the applicant update all other DCD figures, as appropriate, to show any leak off lines in other systems, which were not identified in the response to RAI 235-8275, Question 12.03-45.

In its **response to Question 12.03-52 (ML16133A617)**, the applicant proposed updating DCD Tier 2, Subsection 12.3.1.2.e, "Valves," to state that the application of metallic bellows and diaphragms is limited to valves with low-stroke length applications, or infrequent movement in the APR1400 design. The staff determined this is acceptable, because frequent operation of these type of valves tends to result in leaks.

Also in its response to Question 12.03-52, the applicant indicated that they will use live load packing and double packing with leak off lines for valves greater than 2 inches (5 centimeters [cm]) in diameter and modulating valves and will limit the use of double packing and leak off lines with graphite lantern rings to the valves less than 2 inches (5 cm) in diameter except modulating valves and that all lantern rings used would be graphite lantern rings. The applicant also specified that all valves with stem leakoff were provided in this response. The applicant proposed updates to DCD Section 12.3.1.2, to include this information. This is consistent with EPRI's URD and is an acceptable valve design to reduce leakage.

The applicant also proposed updating DCD Tier 2, Subsection 12.3.1.2, to specify that in COL Information Item 12.3(3), the COL applicant will determine if the SG channel head, including divider plate, pressurizer shell, reactor vessel closure head and bottom head, reactor permanent pool seal, RCS main piping, and J-groove weld surface of the reactor vessel closure head will require electro-polishing or mechanical polishing. As discussed above, electro-polishing and mechanical polishing can significantly reduce the dose rates from components. However, the proposed COL information item did not discuss the potential for polishing some of the other significant components where crud can accumulate, such as the regenerative heat exchanger and refueling pool and refueling cavity liner.

In **Revision 1 of its response to Question 12.03-52 (ML16211A114)**, the applicant included the regenerative heat exchanger, spray valves, refueling pool liner, refueling cavity liner, SFP liner, and cask loading pit liner in the proposed COL 12.3(3), as items the COL applicant should consider for electro or mechanical polishing. It is acceptable for the COL applicant to decide if polishing of these components is required. In addition, in its response, the applicant proposed updating DCD Tier 2, Subsection 3.8.3.6.3, "Stainless Steel Pool Liners," to specify that stainless steel surfaces on the refueling pool conform to a No. 4 finish as specified in American Society for Testing and Materials (ASTM) A480, "Standard Specification for General Requirements for Flat-Rolled Stainless and Heat-Resisting Steel Plate, Sheet, and Strip." This is consistent with the EPRI URD for minimizing time and contamination in decontaminating the refueling pool and is acceptable. The applicant also made other clarifications in Revision 1 of the applicant's response.

The staff determined that the design features and COL information item provided in relation to the applicant's response to Revision 1 of Question 12.03-52, are in accordance with 10 CFR 20.1101(b) and 10 CFR 20.1406 and are, therefore, acceptable. Therefore, RAI 396-8463, **Question 12.03-52, is being tracked as a confirmatory item** until the proposed changes are incorporated into the DCD.

In addition, the design also includes provisions to limit worker dose during resin changes and filter removal. Such as design features to allow the remote removal of filters and work on ion exchangers. One of these design features is that high-activity spent filters, such as filters from the CVCS, can be removed by means of a shielded plug and cask for transfer to the compound building for placement in a 55-gallon (208-liter) drum, as discussed in DCD Tier 2, Section 11.4.2, "System Description." However, the DCD does not provide any information regarding how worker exposure will be maintained ALARA during transfer from the cask to the drum and if there will be any shielding around the drum to limit worker doses. The staff issued **RAI 235-8275, Question 12.03-24 (ML15296A006)**, requesting that the applicant provide this information.

In its **response to Question 12.03-24 (ML16026A562)**, the applicant revised DCD Tier 2, Subsection 11.4.1.5, "Radioactive Source Terms in SWMS," and Subsection 11.4.1.7, "Mobile Equipment," to specify that the high activity filters are removed by a shield plug and a shielded filter handling cask, moved to a filter capping station where they are capped, and shielded while moved to temporary storage. The approach described in the applicant's response and in the associated DCD markups are an acceptable method for reducing worker exposure during the transport of spent filters. Therefore, **RAI 235-8275, Question 12.03-24, is being tracked as a confirmatory item**, pending the incorporation of the proposed DCD changes.

DCD Tier 2, Subsection 12.3.1.2, indicates that tanks are provided with vents to facilitate the removal of potentially radioactive gases. However, the DCD does not indicate where the vent lines send the potentially radioactive gases to or if there are appropriate provisions to ensure the proper operation of the ventilation system. Therefore, the staff issued **RAI 235-8275, Question 12.03-25 (ML15296A006),** requesting that the applicant indicate where vent lines vent radioactive gas to and how the design limits airborne radioactivity. In addition, the staff also requested that the applicant provide information regarding if design features are in place to limit moisture intrusion into the ventilation systems due to the venting of tanks.

In its **response to Question 12.03-25 (ML16036A042)**, the applicant proposed updating DCD Tier 2, Subsection 12.3.1.2, to specify that tanks are vented to the cubicle atmosphere within close proximity to the respective radioactive building HVAC system intake for collection and treatment of radioactive gases. In addition, the proposed DCD update specifies that tanks are provided with internal filters on the vent lines to prevent the release of liquids. This approach limits airborne radioactivity, limits the release of liquid radioactive material, and includes design provisions for preventing the intrusion of liquids into the ventilation system that could potentially damage the ventilation filters, and it is, therefore, acceptable. Therefore, RAI 235-8275, **Question 12.03-25, is being tracked as a confirmatory item**, pending the incorporation of the proposed DCD revisions.

DCD Tier 2, Subsection 12.3.1.2.j, “Refueling equipment,” indicates that refueling equipment is designed to prevent the fuel from being lifted above the minimum safe water depth in order to limit personnel exposure and avoiding fuel damage. However, the application did not indicate the maximum height at which fuel is allowed to be lifted. Therefore, the staff issued **RAI 235-8275, Question 12.03-28 (ML15296A006)**, requesting that the applicant specify the maximum lift height for a fuel assembly in the refueling pool and SFP, considering the highest activity assembly that could be transferred. In this RAI, the staff also requested that the applicant ensure that there is sufficient space for the movement of fuel above the fuel storage racks, given this lift height.

In its **response to Question 12.03-28 (ML15334A454)**, the applicant specified that when a fuel assembly is raised to the maximum lift height in the refueling pool and SFP, there would be a minimum of 2.74 meters (9-feet) of water coverage over the active portion of the fuel, and that mechanical stops would prevent lifting the assembly higher than this level. The applicant also provided additional information and the staff verified, based in part on the response to RAI 98-8051, **Question 09.01.02-8 (ML15299A481)**, that the SFP design allows for fuel to be at that height and have sufficient clearance above the fuel assemblies seated in the pool below. However, the applicant’s revision to DCD Tier 2, Subsection 9.1.4.3, “Safety Evaluation,” indicated that shielding was necessary at this lift height to ensure that the dose to operators on the refueling machine and SFP handling machine platforms is 2.5 mrem/hr (25 µSv/hr) or less in accordance with ANSI/ANS-57.1-1992, “Design Requirements for Light Water Reactor Fuel Handling Systems,” which is referenced by the applicant and in the SRP. Therefore, the staff issued **RAI 396-8463, Question 12.03-50 (ML16034A054)**, requesting that the applicant provide information on the shielding, such as its material, density, thickness, and location and to provide additional information regarding if additional shielding was necessary for workers near the SFP or refueling area but not on the refueling machine and SFP handling machine platforms. The staff also requested that the applicant update the DCD with this information.

In its **revised response to Question 12.03-28 (ML16232A529)**, the applicant removed the language from DCD Tier 2, Subsection 9.1.4.3, which indicated that shielding would be designed later. This information supplements the response to Question 12.03-50, provided below. Therefore, **RAI 235-8275, Question 12.03-28, is being tracked as a confirmatory item**, pending the incorporation of the proposed DCD revisions.

In its **response to Question 12.03-50 (ML16083A547)**, the applicant indicated that the grapple, and a hoist box and mast (only for the refueling pool) is reflected in the shielding analysis, but no additional shielding was necessary. However, the applicant's response did not describe the design of these components or provide any information supporting the statement that these components limit the dose to 2.5 mrem/hr (25 µSv/hr) or less and the staff's confirmatory calculations showed dose rates approximately 10 times greater than the 2.5 mrem/hr (25 µSv/hr) limit when the shielding of these components was not considered.

In **Revision 1 of its response to Question 12.03-50 (ML16232A504)**, the applicant provided detailed design information on the shielding properties of the fuel handling equipment and the methodology for calculating the dose to operators on the refueling equipment and the operating floor of the refueling pool and SFP. The response indicated that for the refueling machine, the grapple, mast, and hoist box are relied upon for shielding; and for the spent fuel handling machine in the SFP, the grapple and hoist box are relied on for shielding. The applicant provided the material specifications, including densities and thicknesses of these components in its response. With the shielding of these components and the 9-feet (2.74 meters) of water shielding, the applicant calculated dose rates of less than 2.5 mrem/hr (25 µSv/hr) at all locations on the fuel handling equipment platforms and the operating floor of both the refueling pool and SFP. The staff reviewed the information provided in the applicant's response and determined that the design of the fuel handling equipment to be acceptable to maintain doses to less than 2.5 mrem/hr (25 µSv/hr) on the trolley machines and the operating floors when handling the maximum spent fuel assembly at 100 hours post shutdown, when considering the 9-feet (2.74 meters) of water shielding and the design of the fuel handling components. Therefore, **the staff closed RAI 396-8463, Question 12.03-50, as resolved**.

The staff also notes that in the applicant's response to **RAI 310-8355, Question 14.03.08-12 (ML15364A593)**, the applicant provided information in DCD Tier 1 indicating that mechanical stops would be provided in order to ensure that fuel assemblies are not lifted higher than 9-feet (2.74 meters) below the water surface, when the shielding provided by the refueling equipment is considered.

DCD Subsection 12.3.1.2.p, "Spent fuel pool (SFP) decontamination," indicates that high-pressure demineralized water will be used to decontaminate equipment that has been in the SFP. However, it is unclear what design features are in place to limit the spread of contamination and the generation of airborne radioactive material when this equipment is being used. The staff issued **RAI 235-8275, Question 12.03-35 (ML15296A006),** requesting that the applicant provide this information.

In its **response to Question 12.03-35 (ML16028A286)**, the applicant proposed updating DCD Tier 2, Subsection 12.3.1.2.p to specify that when high pressure demineralized water is used to decontaminate the SFP or bulky components, protective covers will be used to minimize the spread of contamination and high pressure demineralized water will not be used in open areas above the pool. This response is acceptable for the SFP, however, in a clarification call with the applicant, the staff asked the applicant if the same information would also we applicable to the refueling pool. In **Revision 1 to Question 12.03-35 (ML16124B183)**, the applicant updated

DCD Tier 2, Subsection 12.3.1.2.p to specify that the same criteria are also applicable to the refueling pool. The staff determined that this response is an acceptable approach to minimizing contamination and limiting worker dose. Therefore, **RAI 235-8275, Question 12.03-35, is being tracked as a confirmatory item**, pending the incorporation of the proposed DCD changes.

DCD Tier 2, Section 11.3, "Gaseous Waste Management System," indicates that there are temperature and humidity sensors associated with the gaseous waste management system charcoal delay beds and guard beds. The DCD also indicates that there is a potential that the beds may need to be replaced during the life of the plant. The delay bed rooms are a potential VHRA, which require additional controls beyond those required of normal high radiation areas, as required by 10 CFR 20.1602. Therefore, the staff issued **RAI 225-8254, Question 12.03-13 (ML15268A002)**, requesting that the applicant provide information on when access to the delay beds would be required and discuss how the design limits radiation exposure to workers in accordance with 10 CFR 20.1101(b) and RG 8.8.

In its **response to Question 12.03-13 (ML16018A005)**, the applicant indicated that components such as valves and instruments are not installed in the delay bed and guard bed rooms in order to minimize the radiological exposure to the plant operator. The temperature and humidity instrumentation are installed at wall mounted piping racks outside of the rooms in lower dose rate areas. The applicant also proposed updating DCD Tier 2, Section 11.3.2, "GRS [gaseous radwaste system] Description," to include information on replacing or regenerating the charcoal delay beds, if required. If regeneration is needed the delay bed can be isolated and allowed to regenerate. When replacement is required, the spent charcoal can be quickly removed through the charcoal removal port at the bottom of the bed using the temporary vacuum charcoal removal system at the top of the bed and the fresh charcoal is uniformly loaded into the beds. The radioactive gases in the bed are purged with nitrogen gas before the replacement in order to minimize the radiological exposure to plant workers.

The proposed DCD changes describing the regeneration and replacement of the charcoal are acceptable, however, the applicant never fully answered the staff's question regarding the need to access the delay bed rooms. It is necessary for the staff to review the need to access the delay bed rooms in order to ensure that the design uses engineering controls to the extent practicable to limit radiation exposure, in accordance with 10 CFR 20.1101(b). In a teleconference, the staff requested that the applicant provide additional information regarding the access to the delay bed rooms if it would be needed to work on the physical detectors for temperature and humidity instrumentations or if there would be any other reason to access the delay bed rooms besides potential charcoal replacement. The applicant indicated that it would need to check with the system designer and get back to the staff at a later date. Therefore, RAI **225-8254, Question 12.03-13, is being tracked as an open item** until the applicant provides this additional information.

In addition, DCD Tier 2, Section 11.3.2, discusses the use of the gas stripper to remove dissolved gasses from the reactor coolant. However, the application does not discuss if the possibility of removing gasses directly from the pressurizer to facilitate reactor shutdown is provided in the APR1400 design. Some nuclear power plants have been known to degas the pressurizer directly using the primary sampling system, which could potentially cause equipment degradation issues, and could also potentially result in the unexpected release of radioactive material or an unnecessary worker dose due to the event or system repair, if the primary sampling system is not designed to perform this function. In order to determine if this process of removing radioactive gasses from the pressurizer will be used in the APR1400 design, the staff issued **RAI 235-8275, Question 12.03-43 (ML15296A006),** requesting that the applicant provide information regarding this process if it will be used and if so, whether the primary sampling system equipment is designed for this operation.

In its **response to RAI 235-8275, Question 12.03-43 (ML16057A071)**, the applicant indicated that the APR1400 design does not use the sampling system to degas the pressurizer. Instead in the APR1400, gas is removed from the pressurizer pilot-operated safety relief valves, which is routed to the gas treatment system. The applicant proposed updating DCD Tier 2, Subsection 5.4.10.2.3, "Operation," and Subsection 5.4.12.2.2, "Pressurizer Vent," to provide this information. The staff finds the applicant's response and associated DCD markups are acceptable from a radiation protection perspective because the applicant indicated that the primary sampling system will not be used to degas the pressurizer and the applicant described an alternative means for degassing the pressurizer to facilitate a reactor shutdown.

However, the staff identified an additional question regarding whether similar information should be added to DCD Tier 2, Subsection 5.4.12.2.1, "Reactor Vessel Closure Head Vent," related to the reactor vessel closure head, because the reactor vessel closure head also has the same gas removal function as the pressurizer pilot-operated safety relief valves.

In Revision 1 of its **response to RAI 235-8275, Question 12.03-43 (ML16142A018)**, the applicant removed wording from DCD Tier 2, Subsection 5.4.10.2.3 and edited wording in Subsection 5.4.12.2.2 to provide additional clarification and information on gas removal from the pressurizer. The applicant's revised response is still acceptable from a radiation protection perspective because it described the method for removing gas from the pressurizer and indicated that the sample lines will not be used to degas the pressurizer.

In addition, in its revised response, the applicant proposed updating DCD Tier 2, Subsection 5.4.12.2.1 to describe the gas removal capability of the reactor vessel closure head. The proposed DCD markup indicates that gas can be removed by the reactor vessel closure head using the reactor coolant gas vent piping, which connects to the pressurizer vent and then routed to the reactor coolant gas vent system. However, the reactor systems reviewer indicated that this information is inconsistent with the information already provided elsewhere in DCD Tier 2, Subsection 5.4.12.2.1, which indicates that the piping from the reactor vessel closure head vent goes directly to the reactor coolant gas vent system. The applicant also incorrectly referred to the reactor coolant gas vent system as the reactor coolant vent gas system in the proposed DCD markups. The applicant was made aware of these inconsistencies. Until these inconsistencies are corrected, **RAI 235-8275, Question 12.03-43, is being tracked as an Open Item**.

Section 50.48 of 10 CFR, "Fire protection," requires that the risk of fire-induced radiological hazards to the public, environment, and personnel are minimized and RG 1.189, "Fire Protection for Nuclear Power Plants," states that the plant should maintain the ability to minimize the potential for radioactive releases to the environment in the event of a fire and that radioactive waste buildings, storage areas, and decontamination areas should be separated from other areas of the plant by fire barriers of at least three-hour ratings. The staff reviewed the fire protection design of the APR1400 and determined that the application contained insufficient information for the staff to complete the review. Specifically, numerous areas containing relatively high activity sources were identified as not containing radioactivity. Therefore, it is unclear that potential radioactivity releases were even considered in these areas. As a result, the staff issued **RAI 235-8275, Question 12.03-46 (ML15296A006),** requesting that the applicant provide additional information on preventing the release of radioactive material as a result of fires.

In its **response to RAI 235-8275, Question 12.03-46 (ML16113A462)**, the applicant updated DCD Tier 2, Subsection 9.5A.3.6.4, "F000-RW: Compound Building – Radwaste Area," to provide corrected and updated information regarding fire area F000-RW in the compound building, which includes numerous sources, such as spent resin long-term storage tank and charcoal delay beds. For this area, the applicant removed information indicating that the area is not a radiological area and instead indicated that the area is a radiological area due to the charcoal delay beds and spent resin long-term storage tank. However, the spent resin long-term storage tank is expected to be wet and the resin is enclosed in the tank and there is no ignition source in the spent resin long-term storage tank room. The proposed DCD markup indicates that the charcoal delay beds are normally closed with removable slabs as the only opening to the rooms, which do not allow the propagation of fire.

However, the applicant's response did not acknowledge that there are other radiological sources in fire area F000-RW. In addition, while the applicant's response to **RAI 84-8022, Question 09.05.01-30 (see ML15315A039 for initial response and ML16131A858 for the revised response)** proposed fixing numerous inconsistencies regarding if an area is a radiological area or not, there are still numerous sections in DCD Tier 2, Section 9.5A, "Fire Hazard Analysis," besides fire area F000-RW and those corrected in the applicant's response to RAI 84-8022, Question 09.05.01-30, that contain incorrect or inadequate information regarding the radiological sources in the fire areas. Finally, there are numerous criteria associated with fire protection of radiological material in RG 1.189 that were not addressed in the DCD or in the applicant's responses to Question 09.05.01-30 or Question 12.03-46. These include the prevention of flammable/explosive gasses from ion exchange columns and resins and protecting stored waste from fires and combustibles. The applicant is expected to revise its response to Question 12.03-46, to provide this information. Until the above information is provided, **RAI 235-8275, Question 12.03-46, is being tracked as an Open Item**.

The APR1400 also includes design features to limit access to high and very high radiation areas. For example, the in-core instrumentation chase, is potentially a VHRA during in-core instrumentation withdrawal. A lockable door with a warning light is provided for this area. However, the application does not contain enough information to ensure that appropriate design features are in place to limit personnel exposure and the spread of contamination during the cutting and disposal of in-core instrumentation (such as the use of an underwater cleaning system). In addition, it is unclear that adequate personnel protective features are in place to ensure personnel exposure will be ALARA in the event of AOOs associated with removal of in-core instrumentation. Therefore, the staff issued **RAI 235-8275, Question 12.03-26 (ML15296A006),** requesting that the applicant provide additional information on design features to ensure that the removal of in-core instrumentation is ALARA.

In its **response to Question 12.03-26 (ML16036A042)**, the applicant described the process for withdrawal of in-core instruments (ICIs), disposal, and inserting new ICIs. The APR1400 uses fixed type, bottom mounted, ICIs. During refueling operation, all ICIs are withdrawn from the core when the reactor area is flooded with refueling water. The irradiated portions of the ICIs still remain inside the guide tubes which are filled with refueling water. Since the activities are performed remotely and with the radioactive material shielded underwater, worker exposure is maintained ALARA.

Also, in the applicant's response, it indicated that the cutting of used in-core instrumentation is performed in the refueling pool using a remotely operated hydraulic ICI cutter. The cutting is performed directly above a debris collection funnel so that debris from cutting the in-core assemblies falls into the ICI transport container. The applicant indicated that this minimizes the spread of chips and debris and also minimizes radiation exposure from debris cleanup and consequently, there is no need for an underwater cleaning system to clean refueling pool water during the cutting operation or other refueling activities. However, particles that do not fall into the container would accumulate in the refueling pool water and, when drained, the highly radioactive particles will accumulate on the refueling pool walls and floor. A temporary filtration system would clean the water, leading to a reduction in the dose to workers and the contamination to the refueling pool walls and floor. In addition, it would potentially reduce airborne activity in containment from the particles potentially going airborne. Since the in-core instrumentation would be highly radioactive and a temporary filtration system in the refueling pool could significantly reduce the spread of contamination and the dose to workers, including from radioactivity in the refueling pool beyond particles from in-core instrumentation cutting debris, the staff determined that provisions should be included in the design for the use of a temporary filtration in the refueling pool, to meet the requirements of 10 CFR 20.1101(b) and 10 CFR 20.1406. Therefore, the applicant should provide additional information describing how the design supports the use of a temporary filtration system by providing a power supply capable of supporting a filtration system, if a COL applicant choses to use one. **RAI 235-8275, Question 12.03-26, is being tracked as an Open Item** until the applicant provides this information.

In addition, the application discusses access controls to a few of the high radiation areas in the plant. DCD Tier 2, Table 12.3-5, "Areas Potentially Greater than 1 Gy/hr," provides a listing of all significant high radiation areas (areas with a dose rate greater than 100 rad per hour at 30 centimeters (1 Gray at 1 foot) from the source or surface through which the radiation penetrates) and VHRAs. However, some of these areas do not have any specified access controls. Therefore, the staff issued **RAI 235-8275, Question 12.03-34 (ML15296A006),** requesting that the applicant provide additional information on access controls to other significant radiation areas in the DCD, especially potentially VHRAs, as defined in 10 CFR 20.1602.

In its response to **Question 12.03-34 (ML16028A286)**, the applicant proposed providing additional information regarding controls to significant high radiation areas and VHRAs by adding DCD Tier 2, Subsection 12.3.2.4, "Access Control to High Radiation Area." While the response provided additional information, it was still incomplete, as it did not describe adequate access controls to VHRAs inside containment. RG 8.38 states that each individual VHRA inside containment should be adequately posted and barricaded separately. In addition, the applicant's response did not provide access controls to the fuel transfer tube inspection area, which is a potential VHRA, as identified in Table 12.3-5. The detailed controls to specific areas are summarized below in Revision 2 of the applicant's response to Question 12.03-34.

In **Revision 1 of its response to Question 12.03-34 (ML16124B183)**, the applicant specified additional access controls for VHRAs inside containment and for access to the fuel transfer tube. The applicant also proposed revising DCD Tier 2, Table 12.3-5 to provide additional information on access controls and to clearly distinguish between significant high radiation areas and VHRAs. This approach is acceptable. However, while the instrument calibrator facility (Room 063-P73) is identified in DCD Tier 2, Figure 12.3-10, "Radiation Zones (Normal) Compound Building El. 63'-0"," as an area potentially greater than 500 rad per hour (5 Gy per hour) and Calculation 1-330-N376-009 (reviewed during the shielding audit, discussed below) indicates that it would be a VHRA in the beam of the calibrator, the applicant did not include this area in DCD Tier 2, Table 12.3-5, or provide any access controls to the area.

In **Revision 2 of its response to Question 12.03-34 (ML16174A160)**, the applicant identified the instrument calibrator facility as a VHRA in DCD Tier 2, Table 12.3-5. The instrument calibrator facility has a locked door with a latch bolt operated by key from the outside or by a rotating inside knob/lever, as described in the proposed update to DCD Tier 2, Subsection 12.3.2.4. Additional information regarding access controls and radiation protection design features, especially as they relate to the instrument calibrator facility, are discussed in association with the applicant's response to RAI 376-8496, Question 12.03-49 (below).

The proposed revisions to DCD Tier 2, Subsection 12.3.2.4 provided in the applicant's second revised response to Question 12.03-34 specified that access to the containment building is strictly controlled and the built-in design features which prevent inadvertent access include a secure air lock as the only point of entry for personnel. This door is locked and equipped with a security alarm. In addition to the access control provided at the point of entry into the containment building, separate barriers with individual locked doors are provided for each of the VHRAs. Specific high and very high radiation areas inside containment include the ICI cavity, the hold-up volume tank, the core debris chamber, the reactor cavity, the SG cavity, and the reactor drain tank room.

The proposed revisions to DCD Tier 2, Subsection 12.3.2.4, also specify that the pre-holdup ion exchanger pit, the purification ion exchanger pit, the purification filter pit, and the filter area in the auxiliary building can only be accessed from a manway above these cubicles, which is locked at all times and is under administrative controls to prevent unauthorized access. The volume control tank area is a potential VHRA. This area is locked and can only be opened by key. The manway to the transfer tube inspection area which is a VHRA is provided with a lock.

The proposed revisions to DCD Tier 2, Subsection 12.3.2.4, specify that the significant high and very high radiation areas in the compound building are provided with a latch bolt operated by key from the outside or by a rotating inside knob/lever. The two exceptions are the hot pipe way (on elevation 77') and charcoal delay bed room, which are not provided with doors, are only accessible by hatches above, and are equipped with heavy concrete blocks that make unauthorized access impossible.

In addition, the COL applicant should provide the general operational radiation protection considerations and programs, including a program that conforms with NEI 07-03A, "Generic FSAR [Final Safety Analysis Report] Template Guidance for Radiation Protection Program Description," and NEI 07-08A, "Generic FSAR Template Guidance for Ensuring that Occupational Radiation Exposures are as Low as is Reasonably Achievable (ALARA)," and meets RG 8.38 and other NRC guidance (See COL 12.1(1), 12.1(3), and 12.5(1)). Thus, the applicants must provide adequate procedural controls for VHRAs. Finally, TS Section 5.7, "High Radiation," provides the applicant's alternative approach for complying with 10 CFR 20.1601(a) and 10 CFR 20.1601(b), as allowed by 10 CFR 20.1601(c). TS 5.7 provides access, dosimetry, and surveillance requirements for high radiation areas, such as requiring that access to areas with dose rates greater than 1.0 rem per hour (10 mSv per hour) at 30 centimeters (1 foot) from radiation sources be locked or continuously guarded to prevent unauthorized entry. This TS is consistent with similar TS for other designs and standard TS and is therefore acceptable.

As a result, the staff concludes that the applicant has adequately identified potential VHRAs in the plant and that the design includes adequate features to control access to VHRAs and the other significant radiation areas identified (except for the instrument calibrator facility, which is discussed in more detail below, in the evaluation of the response to RAI 376-8496, Question 12.03-49 (ML16279A511). The design, in conjunction with the radiation protection and ALARA programs provided by the COL applicant and TS 5.7, will ensure that access to high and very high radiation areas are adequately controlled in accordance with 10 CFR 20.1601 and 10 CFR 20.1602. Therefore, the staff finds that Revision 2 of the applicant's response (ML16174A160) to **RAI 235-8275, Question 12.03-34, is acceptable and is being tracked as a confirmatory item**, pending the incorporation of the proposed DCD revisions.

Upon complete resolution of the open items, the staff will determine if the radiation protection facility design features of the APR1400 meets the applicable NRC requirements.

12.3.4.1.2 Shielding

The objective of the plant's radiation shielding is to minimize plant personnel and public exposures to radiation during normal operation (including 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 (see Section 3.11 of this SER for the staff's evaluation for equipment qualification). Shielding is provided to attenuate direct radiation through walls and penetrations and scattered radiation to less than the upper limit of the radiation zone for each area in the DCD Tier 2, Section 12.3, Figures 12.3-1 through 12.3-51, "Radiation zones…," and to ensure that ORE and doses to members of the public remain ALARA. Minimum concrete shielding thicknesses for rooms and cubicles containing significant radiation sources, which require shielding, are provided in DCD Tier 2, Table 12.3-4. These shielding thicknesses are based on the 0.25 percent failed fuel source terms provided in DCD Tier 2, Section 12.2. The applicant used a variety of computer programs to calculate the radiation shielding thicknesses, radiation zoning, and dose rates, as discussed in DCD Tier 2, Subsection 12.3.2.2. Most of the calculations were performed using the Microshield code, however, more advanced computer codes, such as MCNP, "Monte Carlo N-Particle," were used in areas with multiple sources and areas with very high activity sources. The shielding thicknesses provided in Table 12.3-4 are the minimum concrete shielding thicknesses, for each associated room, to which the plant must be designed. The concrete is assumed to have a density of 2.242 grams per cubic centimeter (140 pounds per cubic feet), as specified in the proposed addition to DCD Tier 2, Subsection 12.3.2.2, in **Revision 1 of the applicant's response to RAI 390-8479, Question 12.02-26 (ML16176A372)**. Radiation shielding material other than concrete includes water for spent fuel, control elements, and ICI handled and transferred in the refueling pool and SFP areas, and doors to several radiation areas. These areas that rely on shielding materials other than concrete, are discussed later.

DCD Tier 2, Subsection 12.2.1.6, "Stored Radioactivity," indicates that plant areas storing radioactive wastes are shielded and zoned based on maximum stored waste volumes and average expected source strengths. Since the SRP indicates that source terms used for shielding and zoning should be based on an assumed 0.25 percent fuel failure, the staff issued **RAI 225-8254, Question 12.03-21 (ML15268A002),** requesting the applicant to base the shielding and zoning for storage areas on 0.25 percent fuel failure or justify their approach.

In its **response to Question 12.03-21 (ML16027A265)**, the applicant indicated that the source terms and shielding for the waste drum storage area (low activity storage area) is based on reverse osmosis (R/O) concentrate from liquid waste derived from 0.25 percent fuel failure and dry active waste (DAW) is based on maximum values measured in Korean domestic nuclear power plants. The shielding analysis for the waste drum storage area is based on six months of low activity waste, comprising R/O sludge, spent resin, and other DAW, for a total of 161 drums. Source terms and shielding for the high-integrity container (HIC) storage area is based on the assumed 0.25 percent fuel failure for spent filters and resin. The source terms for the spent filter drum storage and HIC storage area are based on one year of accumulation of waste volumes. The applicant also proposed changes to DCD Tier 2, Subsection 12.2.1.6, consistent with its response. However, while the applicant's response indicated that the waste drum storage area is based on a total of 161 drums. The Calculation 1-330-N376-017 (reviewed as part of the shielding audit, discussed below) indicated that the maximum capacity is 390 drums. In addition, the response did not include enough information on the assumptions used for shielding the HIC storage area. As a result, the response was unacceptable.

In Revision 1 of its **response to Question 12.03-21 (ML16173A248)**, the applicant provided additional information, clarifying that the source terms for the waste drum storage area are based on the R/O concentrate from liquid waste derived from 0.25 percent fuel failure and DAW. The source term for the R/O concentrate liquid waste is conservatively high because the source terms for the R/O concentrate liquid waste, is higher than source terms calculated based on waste from the SWMS.

In addition, the applicant revised the shielding analysis for the waste drum storage area to 326 drums (from 161 drums). The 326 drums exceed the number of drums generated in a six-month period of normal operation, which is expected to be about 290 drums. The six-month of storage is specified in DCD Tier 2, Section 11.4.2. Therefore, the staff determined that the use of six months of storage for the waste drum storage area is acceptable. The 326 drums include 21 drums of R/O concentrate and 305 drums of DAW. The staff confirmed that the 21 drums of concentrate exceeded the amount of R/O concentrate expected to be generated in a year as provided in DCD Tier 2, Table 11.4-1, "Estimated Annual Solid Waste Generation." Adding additional drums to reach the maximum capacity of 390 drums, or to increase the amount of R/O concentrate to the maximum amount that would be generated in six months, in accordance with Table 11.4-1 (approximately 29 drums), would only be expected to minimally increase the amount of radiation shielding needed because of the shielding configuration. For the shielding configuration, the applicant assumed that the higher activity drums are stored in the interior of each layer in the shielding analysis, providing additional shielding for higher activity waste. In addition, the density of the solid waste source contained in the drums was changed from 2.26 grams per cubic centimeter (141.1 pounds per cubic foot) to 1 gram per cubic centimeter (62.4 pounds per cubic foot), which is a conservative value. These changes resulted in the room dose rate increasing from 12.67 mSv per hour (1.267 rem per hour) to approximately 60 mSv per hour (6 rem per hour). However, the original shielding was sufficient and no additional shielding changes were required. The applicant also proposed additional revisions to DCD Tier 2, Subsection 12.2.1.6 to support these changes. The staff concludes that the design includes adequate shielding for the waste drum storage area. A licensee's radiation protection program would also control doses and access to high dose rate areas in the vicinity of the waste drum storage area in accordance with 10 CFR 20.1101(b), if unexpected waste generation rates or operating conditions were to result in a larger than expected source term in the waste drum storage area.

While the response provided adequate information for the waste drum storage area, the response did not provide sufficient information regarding the shielding assumptions for the HIC storage area. In Revision 2 of its response to Question 12.03-21 (ML16189A191), the applicant provided additional information on the shielding for the spent resin storage area, which the staff verified was based on storing approximately 10 years of CVCS spent resin, by comparing the spent resin storage area source term in Calculation 1-330-N376-027 (reviewed as part of the shielding audit, discussed below) to the CVCS ion exchanger source terms in DCD Tier 2, Table 12.2-11, "CVCS Ion Exchanger Inventories, Maximum Values (Bq)." The applicant also proposed adding information to DCD Tier 2, Section 12.2.1.6, to state that the zoning for the waste drum storage area is based on summing the dose rates from the HICs and spent filter drums stored in the room. However, the shielding was based on an assumption that all of the drums are located on one side of the room with HICs on the other side, because the radiation from sources nearest to the wall would be the dominant radiation penetrating the wall. Since the applicant specified this approach in the DCD, and the minimum shield wall thicknesses specified in DCD Tier 2, Table 12.3-4, appear to be consistent with this approach and are reasonable, the staff finds this approach acceptable. As a result of this and other clarifications in Revision 2 of the response, the staff finds the response acceptable. Therefore, **RAI 225-8254, Question 12.03-21, is being tracked as a confirmatory item**, pending the incorporation of the proposed DCD changes.

In addition, DCD Tier 2, Subsection 12.3.2.1 indicates that the shielding provided in the APR1400 design is adequate to maintain dose rates at the exclusion area boundary at less than 25 rem (0.25 Sv) in two hours from direct and scattered radiation during DBAs. However, 10 CFR 100.11, "Determination of exclusion area, low population zone, and population center distance," requires that the total dose from all radiation sources remains below the 25-rem (0.25 Sv) limit in two hours during an accident, which includes gaseous and airborne radioactive material, as well as direct and scatter radiation exposure. Therefore, the staff issued **RAI 235-8275, Question 12.03-31 (ML15296A006),** requesting that the applicant clarify this statement or revise the radiation shielding, as appropriate.

In its **response to Question 12.03-31 (ML16036A042)**, the applicant indicated that the direct dose at 800 meters (2625 feet) from the design basis LOCA source (which is the limiting accident for dose at the exclusion area boundary) is only 2.6 mSv (0.26 rem) and the exclusion area boundary dose from gaseous and airborne radioactive material is 204 mSv (20.4 rem), as specified in DCD Tier 2, Table 15.6.5-14, for a combined dose of 206.6 mSv (20.66 rem), which is less than the 250 mSv (25 rem) limit. In addition, the applicant proposed updating DCD Tier 2, Subsection 12.3.2.1, to clarify that the shielding is adequate so that the total radiation doses to the whole body at the exclusion area boundary do not exceed the 25-rem (0.25 Sv) limit during a two-hour period after a DBA, from all combined radiation sources. This is consistent with 10 CFR 100.11(a)(1) and is acceptable. Therefore, **RAI 235-8275, Question 12.03-31, is being tracked as a confirmatory item**, pending the incorporation of the proposed DCD updates.

The design of the primary shield located around the reactor vessel: 1) limits the radiation level from sources within the reactor vessel and the RCS; 2) minimizes neutron streaming to the containment free volume; 3) minimizes neutron activation of components and structural material; and 4) allows for controlled access for inspections while limiting personnel exposure. However, it is unclear if the applicant appropriately considered potential radiation shielding through piping penetrations in the primary shield wall for workers inspecting RCS piping during an outage and if there are any design features to limit worker exposure from such streaming. The staff issued **RAI 235-8275, Question 12.03-27 (ML15296A006)**, requesting the applicant to provide this information.

In its **response to Question 12.03-27 (ML16057A071)**, the applicant indicated that piping penetrations are located as high above the floor as possible and not in a direct line between a radiation source and an area which may require worker occupancy, wherever practicable, as indicated in DCD Tier 2 Subsection 12.3.1.2.f, “Piping and penetrations.” This includes areas near the primary shield wall. In addition, as discussed in DCD Tier Subsection 12.3.2.3, temporary shielding may be used to limit worker exposure when necessary, during special assignments. These design features in combination with the radiation protection and ALARA programs will ensure that worker exposure near piping penetrations is maintained ALARA, in accordance with 10 CFR 20.1101(b). Therefore, the staff finds the response acceptable and **RAI 235-8275, Question 12.03-27 is resolved and closed**.

The secondary shield that surrounds the RCS equipment, including piping, pumps, pressurizer, and SGs protects personnel from the direct gamma ray radiation resulting from activation and fission products in the RCS fluid. In addition, the secondary shield supplements the primary shield by attenuating neutron and gamma ray radiation escaping from the primary shield. Additional shielded compartments are provided inside containment for several other components, such as the CVCS letdown heat exchanger.

While the shielding is based on the assumed 0.25 percent failed fuel source terms provided in DCD Tier 2, Section 12.2, it was unclear how the applicant determined shielding thicknesses for components which are not modeled in DCD Tier 2, Section 12.2; i.e., shielding for pipes containing radioactive material. Therefore, the staff issued **RAI 14-7858, Question 12.03-4 (ML15142A608)**, requesting that the applicant provide this information and to indicate if any material other than concrete is being relied upon for radiation shielding as part of the permanent plant design. In addition, in Question 12.03-4, the staff asked the applicant to identify any doors or hatches which are being relied on for radiation shielding.

In its **response to RAI 14-7858, Question 12.03-4 (ML15201A377)**, the applicant specified that the only materials being relied on for radiation shielding in the plant are ordinary concrete and a few doors and hatches, including the personnel air lock doors to containment, two equipment hatch doors, and the truck bay doors in the compound building (as well as water shielding for components in the SFP and during an outage, the refueling pool). The applicant proposed to add a new COL Information Item 12.3(1) to DCD Tier 2, Section 12.3, specifying that the COL applicant is to provide the material composition and shielding properties of these doors/hatches, and the thickness equivalent to the minimum required concrete shielding thicknesses. This COL Information Item also indicates that the COL applicant is to provide the service life of the doors/hatches to provide reasonable assurance of functionality throughout the life of the plant. It is appropriate for the doors to provide equivalent shielding as the concrete and it is acceptable for the COL applicant to identify the shielding properties and service life of the doors. Therefore, the proposed new COL information item is acceptable.

In addition to specifying the materials used for shielding, the applicant indicated that shielding for pipe chases and valve rooms are determined based on the number and characteristics of the pipes in the corresponding areas and that the dose rates for pipes are determined using the source term of the corresponding upstream component. Pipes were assumed to be 20-feet (6.1 meters) long and the dose rates were taken at one foot (30 centimeters) away from the pipes. If multiple pipes are present in the room, the dose rates from all pipes were added in the shielding analysis. The response proposed adding this information to the DCD. The staff’s simplified confirmatory calculations using a 20-foot (6.1-meter) long pipe of various diameters for piping used for resin transfer, indicated that there is a potential for dose rates greater than 500 rad per hour (5 Gy per hour) at one foot (30 centimeters) away from these pipes. However, the staff did not find any piping areas running from the purification ion exchanger zoned greater than 500 rad per hour (5 Gy per hour). Also, while RAI 14-7858, Question 12.03-4, requested that the applicant indicate how shielding and zoning was determined for sources which were not explicitly provided in DCD Tier 2, Section 12.2, its response did not provide any information on the shielding and zoning for the ventilation system or ventilation system components. Therefore, **RAI 14-7858, Question 12.03-4, was considered unresolved and closed**, and the staff issued **RAI 263-8329, Question 12.03-47 (ML15296A020)**, requesting that the applicant specify how shielding and zoning was determined for ventilation system components and to provide additional details on how shielding and zoning was determined for piping used to transfer resin from the purification ion exchanger.

In its **response to RAI 263-8329, Question 12.03-47 (ML16050A011)**, the applicant proposed adding a paragraph to DCD Tier 2, Section 12.3.3, "Regulatory Basis," to specify the radiation shielding design and radiation zoning for ventilation system components. The applicant also updated the response to specify the specific rooms that resin transfer lines pass through in the plant and how the resin transfer line source terms were modeled. The applicant indicated that the resin transfer lines were modeled as a pipe filled with 40 percent resin in the pipe.

The staff could not accept the resin transfer line being modeled as 40 percent full of resin for maximum radiation zoning purposes, because operating experience shows that resin transfer lines could potentially clog, which would result in it being essentially 100 percent full of resin. In addition, it was unclear to the staff if some of the piping areas would become potential VHRAs (as defined in 10 CFR 20.1602), when the piping was filled with the 100 percent resin source term. It also appeared that the applicant didn't consider the resin transfer piping running through some of the rooms in the radiation zone drawings, because the zoning in some areas was lower than would be expected if the source term for the resin transfer pipe was considered. Finally, while the proposed DCD updates indicated that duct chases are provided with a minimum of 18-inch (45.7-centimeter) concrete walls for shielding purposes, the response did not provide enough detail on the source term and shielding design for ventilation areas.

In its **revised response to Question 12.03-47 (ML16196A285)**, the applicant recalculated the maximum radiation zoning for the resin transfer lines and ensured that it considered all the rooms in which the resin transfer lines are located. As a result, some of the radiation zones increased during resin transfer operations and the applicant updated the DCD Tier 2, Chapter 12 radiation zoning figures accordingly. It is acceptable for some areas to have increased radiation zoning during resin transfer because resin transfer only occurs on a limited basis and the radiation protection program, as will be provided by the COL applicant, can control the dose to plant personnel and limit access, as appropriate. As a result of the increased source terms, the applicant provided information concluding that none of the areas became a VHRA, as a result of the revised assumptions. The applicant indicated that the maximum dose rate one foot (30 centimeters) away from a resin transfer pipe, assumed to be a straight-lined pipe, 20-foot (6.1-meter) long, was 231 rad per hour (2.31 Gy per hour). Even if the pipe contained a 90-degree bend, the maximum dose rate one-foot (30 centimeters) away from the pipe would be less than 500 rad per hour (5 Gy per hour). This is less than the dose limits for a VHRA of 500 rad per hour (5 Gy per hour) within one-meter (3.3 feet) from the surface of the source. The staff performed a confirmatory calculation one-meter (3.3 feet) away from a hypothetical 90-degree bend in the resin transfer pipe and calculated a dose rate of approximately 150 rad per hour (1.5 Gy per hour), which is less than the 500 rad per hour (5 Gy per hour) dose criteria at one meter (3.3 feet), for a VHRA. Since these areas will not be VHRAs with the assumed 0.25 percent failed fuel source term, no additional physical controls are needed for this area, as specified in 10 CFR 20.1602.

In addition, in its **final, revised response to RAI 343-8420, Question 12.02-24 (ML16174A142)**, item 2, the applicant provided additional information and revisions to DCD Tier

2, Section 12.3.3, explaining how the ventilation systems source terms and their shielding designs were developed. The design includes radiation shielding for ventilation components based on an assumed 0.25 percent failed fuel source terms and one year of accumulation. At least an 18-inch (45.7-centimeter) concrete walls are provided for shielding HVAC duct chases in the auxiliary and compound buildings. The staff finds the information provided in the proposed DCD markups is consistent with RG 8.8 and is, therefore, acceptable. Additional information on the evaluation of source terms for the ventilation systems can be found in Section 12.2 of this SER, related to the response to RAI 343-8420, Question 12.02-24. Therefore, **RAI 263-8329, Question 12.03-47, is being tracked as a confirmatory item**, pending the proposed DCD revisions.

While DCD Tier 2, Table 12.3-4, provides minimum shielding thicknesses for areas requiring shielding, the staff noted that the shielding thicknesses for the refueling canal were not provided. Areas surrounding the refueling canal can become significant high radiation areas or even VHRAs and workers could potentially be overexposed if these areas are not appropriately shielded and appropriate procedures are not in place. Therefore, the staff issued **RAI 22-7930, Question 12.03-6 (ML15155B108)**, requesting that the applicant provide shielding information for the refueling canal, to ensure that zoning around the refueling canal is accurate, and to specify if the shielding design accounts for AOOs.

In its **response to RAI 22-7930, Question 12.03-6 (ML15187A087)**, the applicant indicated that the shielding design considers AOOs and provided some examples of how AOOs were considered in the shielding design, including during purging, fuel handling, maintenance activities, and spent resin transfer. In its response, the applicant also provided shielding and zoning information for the refueling canal. However, the shielding and zoning were based on moving fuel assemblies 100 hours after shutdown. The staff's review of TS 3.9.3, "Containment Penetrations" and TS 3.9.6, "Refueling Water Level," indicate that the TS are based on core alterations being performed 72 hours after a shutdown. Since the TS allow fuel to be moved 72 hours after a shutdown, the staff could not accept the applicant's approach of basing the shielding design for fuel transfer on an assembly being moved 100 hours after shutdown. In addition, the staff's analysis reveals that the spent fuel source term could be significantly higher, possibly more than 30 percent higher, if the fuel is being moved at 72 hours instead of 100 hours. Therefore, the **staff closed RAI 22-7930, Question 12.03-6, as unresolved** and issued **RAI 146-8152, Question 12.02-14 (ML15222B315)**, requesting that the applicant base its shielding and zoning calculations for the transfer of spent fuel 72 hours after operation. As discussed in Section 12.2 of this SER, under the discussion of Question 12.02-14, the applicant proposed updating TS 3.9.6 to specify that the decay time before fuel movement is required to be 100 hours. As a result, it is acceptable to assume that the maximum fuel source term is based on fuel movement 100-hours after shutdown.

However, while the application indicates that access to the transfer tube is provided by the personnel airlock, DCD Tier 2, Table 12.3-4, did not provide minimum shielding thicknesses for the fuel transfer tube and it was unclear if the zoning for areas surrounding the fuel transfer tube was based on the maximum spent fuel assembly passing through the transfer tube. Therefore, the staff issued **RAI 235-8275, Question 12.03-41 (ML15296A006),** requesting the applicant to provide this information.

In its **response to RAI 235-8275, Question 12.03-41 (ML16036A042)**, the applicant specified that the fuel transfer tube area is surrounded by the plant north and south walls, while the west side adjoins to the containment, and the east side connects to the refueling canal. The areas above and below the refueling canal are pipe chases. The applicant also noted that the transfer tube inspection area (Room 113-A01B) is not to be accessed during refueling operations and is accessed from a hatch above. The applicant also provided the shielding thicknesses for the transfer tube area in DCD Tier 2, Table 12.3-4. The transfer tube has wall thicknesses of 44 inches (1.12 meters) to the north and south, 62 inches (1.57 meters) for the floor, and 60 inches (1.52 meters) for the ceiling. There are no walls on the east side (refueling canal) or west side (refueling pool). However, the transfer tube inspection area (Room 113-A01B) was not labeled on the Chapter 12 radiation zone figures and there were inconsistencies in the room name title given to the transfer tube access area (137-A40B), within the response and the DCD.

In **Revision 1 of its response to Question 12.03-41 (ML16116A428)**, the applicant corrected the inconsistencies in the room names and labeled Room 113-A01B as the transfer tube inspection area. The staff performed confirmatory Microshield calculations of the maximum fuel assembly, with a peaking factor of 1.55, being transferred through the refueling canal. The results showed dose rates of approximately 365 mrem (3.65 mSv) per hour through the 44-inch (1.12 meter) wall at one foot (30 centimeters) away from the walls); 1.8 mrem (18 μSv) per hour through the 60-inch (1.52-meter) ceiling (at one-foot (30 centimeters) away from the floor of the room above); and 1.1 mrem (11 μSv) per hour through the 62-inch (1.57-meter) floor (at one-foot (30 centimeters) away from the ceiling of the room below). These results should be doubled to conservatively account for the two bundles that can be transferred through the transfer tube, at one time, in the APR1400 design. This is consistent with the radiation zone designation of less than 1 rem (10 mSv) per hour for the rooms north (120-A16B) and south (120-A16A) of the transfer tube shown on DCD Tier 2, Figure 12.3-5, "Radiation Zones (Normal) Auxiliary/Containment Building El. 120'-0"," and 1 rem (10 mSv) per hour below the fuel transfer canal (078-A21B) shown on DCD Tier 2, Figure 12.3-4, "Radiation Zones (Normal) Auxiliary/Containment Building El. 100'-0"." Room 137-A40B, above, was labeled less than 2.5 mrem (25 μSv) per hour, which is slightly lower than the conservative staff's calculated value of 3.6 mrem (36 μSv) per hour for two assemblies. However, in Revision 1 of its response to Question 12.03-41, the applicant included a note on DCD Tier 2, Figure 12.3-6, "Radiation Zones (Normal) Auxiliary/Containment Building El. 137'-6"," indicating that the dose rate in this room could be Zone 3 (less than 5 mrem (50 μSv) per hour) during refueling operations, based on their own calculations. Therefore, the staff finds that the shielding design and radiation zoning for the fuel transfer tube are acceptable.

In **Revision 2 of its response to Question 12.03-41 (ML16174A160)**, the applicant also provided additional notes to DCD Tier 2, Figure 12.3-5, to specify that the refueling pool area (130-C01), transfer tube inspection area (113-A01B), and refueling canal (119-A01B) are radiation Zone 8 (greater than 500 rad (5 Gy) per hour), during fuel transfer. These areas are underwater during fuel transfer and it is appropriate to label them as Zone 8, when fuel is present. Therefore, the applicant's response to **RAI 235-8275, Question 12.03-41, is being tracked as a confirmatory item**, pending the proposed DCD revisions provided in Revision 2 of the applicant's response.

The primary direct radiation hazard in the RCS during operation is due to N-16. Since the half-life of N-16 is 7.13 seconds, N-16 is generally not a significant radiological hazard outside containment in large PWRs, because of a sufficiently long transit time from the core to when the fluid leaves containment. However, it is unclear if the applicant appropriately considers N-16 in CVCS piping immediately after exiting containment in the shielding and zoning design. Likewise, it was unclear to the staff if RCS sample lines appropriately included provisions to allow for N-16 decay. The staff issued **RAI 225-8254 Question 12.03-22 (ML15268A002)**, requesting that the applicant provide additional information regarding N-16 concentrations in piping that exits the containment and to ensure that the shielding and zoning design appropriately considers N-16. In its **response to RAI 225-8254, Question 12.03-22 (ML16029A042)**, the applicant indicated that the transit time of the RCS fluid from the core to the CVCS letdown line containment penetration is assumed to be 62.43 seconds, and calculated an N-16 concentration of 1.94E+04 Bq per gram (0.524 micro curie per gram), based on this delay time. The applicant proposed adding this information to DCD Tier 2, Table 12.2-7, "N-16 Activity," along with a note explaining how these values were estimated. As a result of the 62.43 second delay time, N-16 is not a major contributor to the dose rates in the auxiliary building.

The 62.43 second delay time frame is based on an assumed transit time from the core to the midpoint of the SG of 3.85 seconds, the residence time through the regenerative heat exchanger of 4.95 seconds, and residence time through the letdown heat exchanger of 53.63 seconds. The applicant's calculation conservatively did not account for the transit time from the midpoint of the SG to the regenerative heat exchanger, the transit time between the regenerative heat exchanger and the letdown heat exchanger, or the transit time from the letdown heat exchanger to where the piping exits containment. Although not accounted for in the calculation, in **Revision 1 of the applicant's response to Question 12.03-22 (ML16116A374)**, the applicant proposed including information in DCD Tier 2, Table 12.2-7, indicating that there will be at least 200 feet (61 meters) of piping from the letdown heat exchanger to the containment penetration.

The 3.85 second transit time from the core to the midpoint of the SG specified in DCD Tier 2, Table 12.2-7, is consistent with values provided with other previously approved similar designs. Using the average length of the U-tubes (but not accounting for the curvature of the tubes or the plenum portion of the tube side of the heat exchangers before and after the tubes), the staff calculated an average transit time of 3.5 seconds through the regenerative heat exchanger and 22.6 seconds through the letdown heat exchanger (verses the 4.95 seconds and 53.63 seconds, values indicated by the applicant). In addition, while the applicant did not consider the piping from the letdown heat exchanger to the containment penetration area, using the letdown line piping diameter of 2 inches (5.1 centimeters) (specified in DCD Tier 2, Subsection 5A.4.9.4.2, "Design Evaluation") and a normal letdown flow rate of 80 gallons per minute (303 liters per minute) (specified in DCD Tier 2, Table 9.3.4-3, "Chemical and Volume Control System Parameters"), the average transit time through the 200 foot (61 meter) section of piping would be approximately 24.5 seconds. Based on this, the average transit time from the core to the containment penetration area would then be 54.5 seconds (3.85 + 3.5 + 22.6 + 24.5). Since, this does not consider the transport time between the midpoint of the SG and the regenerative heat exchanger, the piping between the heat exchangers, the plenum area of the heat exchangers, or the curvature of the tubes, the transit time would be higher than this. Also, in reviewing the figures of containment shown in DCD Tier 1, Subsection 2.2.1.1, "Design Description," and Tier 2 Section 12.3 and Figure 5.1.2-1, "Reactor Coolant System Flow Diagram," which shows the letdown line between the SG and the reactor coolant pump, the staff concludes that there must be at least 50 feet (15.24 meters) of piping between the start of the letdown line to the regenerative heat exchanger, which would add approximately six more seconds to the transit time. In addition, from looking at the figures, the staff concludes that there must be at least 20 feet (6.1 meters) of piping between the regenerative heat exchanger and the letdown heat exchanger, which would add approximately 2.5 more seconds. Therefore, when the entire flow path is considered, the staff can conclude that there would be an average of at least 63 seconds of delay between the core and the letdown line exiting containment when the 200 foot (61 meter) pipe length from the letdown heat exchanger to the penetration area is considered. Since this does not consider the plenum region of the heat exchangers or the curvature of the tubes, the average transport time would likely be more than 63 seconds, which would result in less N-16 and less radiation dose than what the applicant calculated (based on 62.43 seconds). As a result, the staff can conclude that the radiation zoning and radiation shielding design appropriately accounts for N-16 and the design reduces CVCS N-16 concentrations to acceptable levels outside of containment.

In **Revision 1 of its response to Question 12.03-22 (ML16116A374)**, the applicant also indicated that the RCS sampling lines include delay lines and/or sufficient piping lengths to allow for more than 60 seconds for N-16 decay. This is consistent with DCD Tier 2, Section 9.3.2, "Process and Post-Accident Sampling System," which specifies that the sample lines allow for at least 60 seconds of N-16 decay before exiting the containment building. This minimizes the dose from N-16 to workers and is acceptable. Based on the above, the staff determined that this response is acceptable, pending the proposed DCD changes specified in Revision 1 of the applicant's response. Therefore, **RAI 225-8254, Question 12.03-22, is being tracked as a confirmatory item**, pending the incorporation of the proposed DCD revisions**.**

In reviewing the radiation shielding outside containment, the staff also identified several other rooms and cubicles which contain potentially significant radiation sources for which minimum shielding thicknesses were not provided in DCD Tier 2, Table 12.3-4, including the spent filter drum storage area and the primary sampling sink room. In addition, the staff identified a room number discrepancy and needed additional clarification on which rooms gaseous waste management system charcoal delay beds were located. Therefore, the staff issued **RAI 141-8098, Question 12.03-8 (ML15221A003)**, requesting the applicant to provide this information.

In its **response to Question 12.03-8 (ML15308A572)**, the applicant provided the missing shielding information in DCD Tier 2, Table 12.3-4, and provided justification for not providing shielding for a few of the areas requested. The applicant also explained that the delay beds are located in Rooms 096-P01 and 096-P02, with the first two beds in Room 096-P02 and the second two beds in Room 096-P01. In addition, the applicant provided shielding for other areas that were not included in Table 12.3-4. Finally, the applicant indicated that the minimum radiation shielding information in DCD Tier 2, Table 12.3-4, did not consider the shielding that is required to provide access to vital areas, and identified other errors in the values provided in Table 12.3-4. The response included over 200 revisions to the shielding thicknesses provided in the tables. In order to evaluate these changes, the staff conducted an audit of the applicant's shielding calculations (see ML16036A024 for the audit plan). During the audit, the staff reviewed the applicant's calculations for the radiation shielding and zoning of selected areas for normal operation and vital area accessed during accidents, as well as calculations for determining the radiological effects on equipment important to safety as discussed in Section 3.11 of this SER. As part of the audit, the staff asked the applicant to provide information justifying the use of different dose conversion factors for piping areas than were used for all other radiation dose and shielding calculations in the APR1400 design. Specifically, the applicant used the dose conversion factors in International Commission on Radiological Protection Publication 74 (ICRP 74), "Conversion Coefficients for Use in Radiological Protection Against External Radiation," for the piping calculations while ICRP 51, "Data for Use in Protection against External Radiation," was used for all other calculations. The ICRP 74 dose conversion factors would result in a lower dose rate from the piping than ICRP 51. The applicant also did not consider the effects of radiation backscatter in the piping calculations and did not provide any justification for not considering backscatter in the calculations. Therefore, the audit remains open until the applicant responds to the staff's questions. Additional details on the shielding audit will be provided in the audit report when it's completed. Therefore, **RAI 490-8599, Question 12.03-53 (discussed below), was issued as a result of the shielding audit**, and **RAI 141-8098, Question 12.03-8, remains an open item** until the shielding audit items are resolved. In addition, the staff notes that several other RAI responses also proposed changes to DCD Tier 2, Table 12.3-4. The final revision of DCD Tier 2, Table 12.3-4, will be reviewed in detail to ensure that all proposed changes are correctly incorporated.

In reviewing the radiation shielding and zoning for the compound building, there was insufficient information in the application to determine if the shielding and zoning for Room 077-P01 (hot pipe way) in DCD Tier 2, Figure 12.3-11, "Radiation Zones (Normal) Compound Building Partial Plan," was adequate. The room includes piping used to transfer highly radioactive fluid and encompasses much of the 77-foot elevation of the compound building. It is identified as an area that could potentially exceed 100 rad (1 Gy) per hour. It is an abnormally shaped room with many different walls shielding numerous areas. In addition, there is a stairwell and elevator, which run up through the center of the piping area and are required to be shielded from the piping area on all sides. The information provided in DCD Tier 2, Table 12.3-4, is insufficient to determine the shielding thickness for each of the walls in the room, including for the stairwell and elevator that run through the room. Therefore, the staff issued **RAI 141-8098, Question 12.03-10 (ML15221A003),** requesting the applicant to provide additional information regarding the shielding for this room.

In its **response to Question 12.03-10 (ML15308A572)**, the applicant provided a figure showing the radiation shielding information for Room 077-P01 but did not include the shielding information in the DCD. In its **revised response to Question 12.03-10 (ML16201A239)**, the applicant proposed including some of the shielding information for Room 077-P01 in the DCD. In addition, the applicant proposed including some of the shielding information in Room 068-A07A (hot pipe way) and Room 068-A10A (filter and demineralizer valve room), which are irregularly shaped and significant, locked high radiation areas in the auxiliary building. The staff evaluated the shielding information provided and found it appropriate for the radiation sources in those rooms. However, the thickness for many of the shields were still not provided in the proposed DCD update. The applicant is expected to provide the missing information in a second revision to its response. Therefore, RAI **141-8098, Question 12.03-10, is being tracked as an Open Item** until the applicant provides all shielding thicknesses for these areas in the DCD.

The staff also requested the applicant to justify shielding and zoning information and provide additional miscellaneous information associated with the review of the Chapter 12 radiation zone maps in **RAI 141-8098, Question 12.03-9 (ML15221A003)**. Information requested included justifying specific shield wall thicknesses, zoning for specific rooms, access to certain locations, and other clarifications related to the radiation zone figures, in order to ensure that the design limits radiation exposure to ALARA. Specifically, some of the information requested by the staff was for the applicant to provide: 1) justification for shielding thicknesses for the instrument calibrator facility, which is identified as a potential VHRA in the beam of the unexposed calibration radiation source; 2) the shielding thicknesses around the pressurizer, SGs, and reactor coolant pumps in DCD Tier 2, Table 12.3-4; and 3) information as to whether containment entries are expected to be made during operation.

In its **response to Question 12.03-9 (ML15308A572)**, the applicant indicated that the calibrator facility shielding was based on the expected orientation of the instrument calibrator. Since the east wall is facing in the opposite direction of the calibrator source beam, it gets much less radiation exposure than the other walls and requires less shielding (18 inches [45.72 centimeters]). Since the west wall gets direct exposure to the calibrator beam, more shielding is required (48 inches [1.22 meter]). As a result, the shielding for the instrument calibrator facility is acceptable. The applicant also, provided the shielding thicknesses for various components inside containment in Table 12.3-4, and updated DCD Tier 2, Section 12.3.1, to specify that limited access to the annulus areas of containment outside the secondary shield walls are allowed during operation with limited occupancy with administrative entry controls. The staff finds this approach is consistent with RG 8.8 and is, therefore, acceptable. The applicant also addressed the staff's other questions in RAI 141-8098, Question 12.03-9, including providing information on the designations for removable shield walls and labyrinths on the plant figures. However, the applicant's response and information in the DCD did not provide any information regarding the shielding attenuation capabilities of the removable shield walls and labyrinths. In addition, while the applicant provided some of the information regarding shielding in containment, it did not provide any information for the west side of the primary shield wall (below 114'-6" elevation) in DCD Tier 1, Table 2.2.1-1, "Definition of Wall Thicknesses for Nuclear Island Structure," which already contained the thicknesses of all the other primary shield walls.

In its **revised response to Question 12.03-9 (ML16144A675)**, the applicant provided the shielding thickness for the west side of the primary shield wall below 114'-6" elevation, which is 6'-7" (2 meters), and the east and west side from 114'-6" to 130' elevation, which is also 6'-7" (2 meters)." These thicknesses are comparable to the other primary shield walls. Since access to areas around the primary shield wall are not expected during operation and access to these areas would be administratively controlled under the radiation protection program, this thickness is acceptable. In its revised response, the applicant also proposed updating DCD Tier 2, Subsection 12.3.1.1, to specify that labyrinths are provided at the entrances to high radiation areas and that the scattered dose rate through the passageway and the transmitted dose rate through the shield wall from all contributing sources will be below the upper limit of the radiation zone specified for each area. The staff finds this design is consistent with an ALARA design and is acceptable. In Revision 2, to the applicant's response to Question 12.03-09 (ML16175A687), the applicant specified that the plants removable shielding is designed to limit the radiation level so as not to exceed the radiation zone boundary in the surrounding areas and that its density is greater than the permanent concrete shield wall and that the gap with the concrete structure is designed to be less than 0.5 inches (1.26 centimeters) with an offset. The use of removable shielding that limits radiation exposure as described above is consistent with guidance in RG 8.8; the removable shielding allows easy access and removal of components, if necessary, thus limiting radiation exposure to workers. Since the removable shielding will be designed to limit dose rates equivalent to that of the permanent plant shielding, the staff finds the removable shielding design and proposed DCD changes acceptable. As a result, Revision 2 of the response to **RAI 141-8098, Question 12.03-9 is being tracked as a confirmatory item**, pending the proposed DCD changes.

In DCD Tier 2, Chapter 11, COL Information Item 11.4(7) indicates that the COL applicant is responsible for the provisions of an interim radwaste storage facility, if necessary. Therefore, the staff issued **RAI 235-8275, Question 12.03-38 (ML15296A006),** requesting that the applicant update this COL information item or provide a new COL information item to ensure that the shielding and zoning for an interim radwaste storage facility are in accordance with 10 CFR 20 and the guidance in SRP Section 12.3-12.4. In its **response to Question 12.03-38 (ML16028A286)**, the applicant proposed updating COL Information Item 11.4(7) to specify that the COL applicant must also ensure that the design and operation of the facility meets the requirements of 10 CFR 20 and the guidance of RG 4.21, RG 8.8, and NUREG-0800, Section 12.3-12.4, if they include such a facility. Since the COL applicant would be required to address all applicable requirements regarding the design and operation of this facility, the applicant's response is acceptable. Therefore, **RAI 235-8275, Question 12.03-38, is being tracked as a confirmatory item**, pending the incorporation of the proposed DCD revisions.

Section 50.34(f)(2)(vii) of 10 CFR, requires that vital areas which require access by operators aiding, mitigating or recovering from the accident, need to be identified. NUREG-0737, section II.B.2, "Plant Shielding," provides additional guidance for meeting 10 CFR 50.34(f)(2)(vii). The source terms used for the shielding design for post-accident sources are discussed in Section 12.2 of this SER. DCD Tier 2, Subsection 12.3.1.8, "Post-Accident Radiation Zones," and Subsection 12.3.1.9, "Vital Area Access," discuss post-accident radiation zoning and vital area access. Subsection 12.4.1.2, "Post-Accident Actions," discusses mission dose rates for performing vital functions and DCD Tier 2, Table 12.4-8, "Estimated Accident Mission Dose," provides the mission dose rates. Subsection 12.3.1.8, and Subsection 12.3.1.9, indicate that the MCR and TSC require continuous access and that the post-accident sampling system, remote shutdown room, and remote control console room, Class 1E switchgear room, instrumentation and control (I&C) equipment room, and access areas outside the containment spray, and shutdown cooling pump rooms require limited access and that the resulting dose to access and perform vital actions in these areas will not exceed 5 rem (50 mSv). The DCD also indicates that the safety injection system, containment spray system, shutdown cooling system, and CVCS system were considered for post-accident access but do not constitute vital areas. However, it is unclear why these areas would not constitute vital areas. Therefore, the staff issued **RAI 235-8275, Question 12.03-30 (ML15296A006),** requesting that the applicant provide justification for why these areas are not vital areas. In addition, in this RAI, the staff requested additional information regarding vital areas, including indication of what vital actions are performed outside the shutdown cooling and containment spray pump rooms and to request the applicant to clarify if DBAs other than the design-basis LOCA were considered for vital area access.

In its **response to RAI 235-8275, Question 12.03-30 (ML16028A286)**, the applicant specified that while the safety injection system, containment spray system, and shutdown cooling perform important safety functions in the event of an accident, there is no need for operators to access these areas during an accident because the systems remotely start when needed and can be controlled from the MCR and the remote shutdown room. However, the areas outside the containment spray and shutdown cooling pump rooms may be required to be accessed following an accident in order to allow interconnection between the shutdown cooling line and the containment spray line which allows the shutdown cooling pump to perform containment spray function. Therefore, these areas are considered to be vital areas in the APR1400 design (and the mission dose to these areas has already been calculated in the DCD). In addition, the applicant indicated that the CVCS is not required to perform any accident mitigation or safe-shutdown function. Accordingly, accessing the CVCS is not considered to be a vital function. The applicant's response included proposed DCD updates that provide this information.

The response also specified that the emergency diesel generators and associated oil system and fuel oil tanks do not need to be accessed during an accident and that no operator action is needed to keep a steady supply of fuel oil to the emergency diesel generators to safely shutdown the plant and maintain a safe-shutdown. Therefore, these areas are not considered vital areas. In addition, two of the diesel fuel storage tanks are located in the auxiliary building and another two are located in the emergency diesel generator building. The two fuel storage tanks in the emergency diesel generator building would not be expected to have high levels of radioactivity during an accident. Based on the above, the staff determined that it is acceptable for these areas to not be considered vital areas.

Finally, in its response to Question 12.03-30 (ML16028A286), the applicant also specified that while other DBAs were considered in determining post-accident source terms and dose assessment, the LOCA was the bounding accident for radiation exposure and that the accident analyses and dose assessments were performed in accordance with the guidance in RG 1.183 and NUREG-0800. The response also proposed updating the DCD to specify that post-LOCA conditions are bounding. A review of Chapter 15 shows that the doses in the MCR and TSC are highest following a LOCA. In addition, release rates are highest during a LOCA. Furthermore, the dose in the safety systems which recirculate post-accident fluid would be highest during a LOCA. Accordingly, the dose to most plant areas would be highest during a LOCA event. In reviewing the vital areas and access routes, the staff agrees that the LOCA would result in the highest doses for performing vital missions. Therefore, the applicant's approach is acceptable. As a result of the above, the applicant's response to **RAI 235-8275, Question 12.03-30, is acceptable and is being tracked as a confirmatory item**, pending the proposed DCD updates.

The staff reviewed the shielding design for access to vital areas and mission dose rates as part of the shielding audit and in reviewing the applicant's revised response to **RAI 141-8098, Question 12.03-8 (ML17012A374)**, and other RAI responses which revised the minimum required shielding thicknesses provided in DCD Tier 2, Table 12.3-4 (discussed above). The staff determined that they are based on the design basis source terms and are generally acceptable. However, as part of the shielding audit, the staff noted that the shielding calculations for determining the dose to operators performing vital missions were based on the use of the Microshield computer code. The Microshield computer code does not allow for modeling complex geometries and does not have the accuracy of more complex shielding codes such as the MCNP computer code. A few of the mission dose rates in DCD Tier 2, Table 12.4-8, are near the 5 rem (50 mSv) limit (for example the dose to perform mission functions in the remote control console room two hours after a LOCA is 4.788 rem (47.88 mSv) and there does not appear to be any extra conservatism built into the calculations. In addition, during the shielding audit, the staff noted a small error in the source term used for post-accident recirculating fluids that resulted in the source term being low by approximately two percent, which in turn would result in calculated doses being even closer to the 5 rem (50 mSv) limit. Finally, there were numerous RAIs and source term changes related to questions discussed in Section 12.2 of this SER. To ensure that the combination of source term changes, the correction of errors, and the limitations of the Microshield computer code does not result in the mission dose to workers exceeding 5 rem (50 mSv), the staff issued **RAI 490-8599, Question 12.03-53 (ML16146A015)**. This RAI requested the applicant to consider the cumulative effects of these changes and the limitations of the Microshield computer code in ensuring that the mission dose rates remain acceptable. In Question 12.03-53, the staff also requested that the applicant consider the cumulative effects of changes in the general radiation shielding design. A response to this question has not been received. Therefore, **RAI 490-8599, Question 12.03-53, is being tracked as an Open Item.**

The MCR is designed to limit the dose to control room personnel to less than a 5-rem (50 mSv) whole body dose for the duration of a DBA, in accordance with GDC 19. A discussion of the shielding of direct radiation sources to the control room can be found in Subsection 12.2.4.3," of this SER, and a comprehensive review of compliance with GDC 19 can be found in Chapter 15 of this SER.

Upon complete resolution of the open items, the staff will determine if the radiation shielding design of the APR1400 meets the applicable NRC requirements.

12.3.4.1.3 Ventilation

RG 8.8 and SRP Section 12.3-12.4, contain guidance on acceptable ventilation design features to control airborne radioactivity levels and maintain personnel doses ALARA. The ventilation systems are designed to ensure that personnel exposure to airborne radioactivity levels is minimized and maintained ALARA and within the applicable limits of 10 CFR Part 20.

In general, in the APR1400 design, ventilation pathways in radiologically controlled areas flow from areas anticipated to have lower levels of airborne activity to areas anticipated to contain higher levels of radioactivity. The ventilation systems are anticipated to provide sufficient flow to maintain airborne activity levels outside containment during operation and inside containment during refueling outages to less than 1 DAC, based on estimated leak rates from components and on design specifications. Minimum design ventilation flow rates for individual cubicles anticipated to contain airborne radioactivity are provided in DCD Tier 2, Table 12.2-26. RAIs related to determining airborne activity concentrations and ventilation system flowrates are discussed in Section 12.2 of this SER.

During accident conditions, the MCR emergency ventilation system is designed to maintain doses in the control room to less than 5 rem (50 mSv) during the duration of a DBA in accordance with 10 CFR Part 50, GDC 19. Section 12.2 of this SER discusses the source terms and shielding for the MCR emergency filters, during accident conditions (see the discussions related to RAI 207-8247, Question 12.02-16, and RAI 390-8479, Question 12.02-26).

Ventilation systems are also discussed in Section 6.4, “Habitability Systems,” and Section 9.4, “Heating, Ventilation and Air Conditioning Systems,” of this SER.

12.3.4.1.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

The area radiation monitors are discussed in DCD Tier 2, Section 12.3.4, and are provided in DCD Tier 2, Table 12.3-6. The containment upper operating monitors (RE-233A and RE-234B) are high range safety-related monitors required by 10 CFR 50.34(f)(2)(xvii) to monitor radiological conditions in containment during an accident. These monitors meet the range and placement criteria specified in NUREG-0737, Item II.F.1.3, “Containment high-range monitor.” The containment (lower) operating area monitors (RE-231A and RE-232) are safety-related monitors in place in case of a fuel handling accident. In addition to their monitoring function, these four containment monitors are also designed to initiate the safety-function of containment purge isolation in the event high radiation levels are detected. The SFP area monitors are also safety-related and have a safety function of initiating fuel handling area emergency ventilation upon detection of high radiation levels in the SFP area. All other area radiation monitors in the APR1400 design are non-safety-related.

All plant area radiation monitoring equipment is designed to alert 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 DCD 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 and the guidance of SRP Section 12.3-12.4. The radiation monitors consist of ionization chambers or Geiger-Mueller tubes.

Radiation levels can be read locally or in the MCR. Alarms are also provided locally and in the MCR. Area monitors are located based on the expected frequency of access, occupancy time, and expected and potential radiation levels in plant work areas. Areas that are typically high-radiation areas but require little or no access (e.g., pipe chases) are not provided with an area radiation monitor. However, areas that typically have a high frequency of access and are normally low-radiation areas, but are potentially high-radiation areas, are provided with an area radiation monitor. In accident conditions, areas such as corridors outside personnel and equipment hatches may become VHRAs and some of these areas are provided with an area radiation monitor to provide reasonable assurance of worker safety. The area radiation monitors are provided in DCD Tier 2, Table 12.3-6, and the locations of area and airborne radiation monitors are provided in DCD Tier 2, Figure 11.5-2A through Figure 11.5-2BB. The guidance of ANSI/ANS-HPSSC-6.8.1-1981, which is referenced in SRP Section 12.3-12.4 and in the DCD, provides examples of appropriate locations for radiation monitors in PWRs. Some of these monitors are not provided in the APR1400 design. Examples include the radwaste and drumming station control panel areas and safe-shutdown control panel areas. Therefore, the staff issued **RAI 225-8254, Question 12.03-20 (ML15268A002),** requesting that the applicant provide these monitors or justify why they are not needed in the APR1400 design.

In its **response to RAI 225-8254, Question 12.03-20 (ML16027A265)**, the applicant proposed updating COL Information Item 11.4(3), to specify that the COL applicant is to prepare a plan to develop and use operating procedures and portable radiation monitoring instruments to ensure requirements and guidance are met for mobile and temporary solid radwaste processing equipment, including those for drumming and capping operations. The applicant also points out that drumming and capping can be done remotely, therefore permanent radiation monitors are not required. Since the shutdown system pump and heat exchanger is normally a locked radiation area, it is not necessary to have permanent radiation monitors in this area. Also, the auxiliary building and compound building HVAC filter areas do not contain area radiation monitors. However, there are effluent monitors downstream of the filters which continually monitor effluent releases. Since these monitors would notify operators of a significant increase in the radioactivity in the HVAC system, area radiation monitors are not required in these areas. Finally, the remote shutdown panel room is located in a clean area of the plant that is well shielded from radiation sources, including during an accident. Therefore, there should not be significant radiation levels in that area of the plant during normal operation or accident conditions. As a result, a radiation monitor is not provided in this area. In addition, the applicant submitted **Revision 1 of its response to Question 12.03-20 (ML16142A027)**, to correct an editorial error. Based on the above, the staff finds the response to **RAI 225-8254, Question 12.03-20, acceptable and this question is being tracked as a confirmatory item**, pending the incorporation of the proposed DCD changes.

As discussed above, the containment operating area, containment upper operating area, and SFP area monitors are safety-related, Seismic Category 1 monitors, connected to a Class 1E power supply that provides safety-related functions of isolating containment purge and initiating emergency ventilation in the SFP area. This is consistent with requirements of 10 CFR 50.34(f)(2) and the guidance of NUREG-0737 and RG 1.97. However, RG 1.97 references Institute of Electrical and Electronics Engineers Standard (IEEE Std.) 497-2002, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations," which provides criteria for classifying equipment needed during DBAs. Based on this guidance, and since the main steam line area monitors (provided in DCD Tier 2, Table 11.5-1, "Gaseous Process and

Effluent Radiation Monitors," instead of Table 12.3-6, because their primary purpose is to prevent potential effluent releases and notify operators of primary to secondary leakage, instead of to protect workers) are designed to detect primary to secondary leakage, such as during a SG tube rupture, the staff issued **RAI 225-8254, Question 12.03-14 (ML15268A002),** requesting that the applicant justify the classification of the main steam line area radiation monitors as non-safety-related, non-class 1E, and Non-Seismic Category 1 monitors.

In its **response to Question 12.03-14 (ML15365A240)**, the applicant referenced its response to RAI 116-8054, Question 14.03.08-5 (ML15303A426), which specified that the main steam line monitors (RE-217, RE-218, RE-219, and RE-220) are mounted on the main steam lines and are designed to detect primary to secondary leakage during normal operation to ensure that SG leakage does not exceed 150 gallons per day (568 liters per day) per SG as required by TS 3.4.12 and to detect a SG leak rate of 30 gallons per day (114 liters per day) as specified in NEI 97-06, "Steam Generator Program Guidelines." The applicant also specified that other safety-related systems provide accident monitoring indications for a SG tube rupture. These indications are RCS pressure, pressurizer level, and SG level. Therefore, the main steam line monitors are not required to be safety-related for the purposes of identifying, monitoring, or automatically initiating safety systems in the event of a SG tube rupture. Based on the above, the staff finds the response acceptable and **RAI 225-8254, Question 12.03-14, is resolved and closed**. Radiation monitoring for the purposes of detecting releases of radioactive material in the event of SG tube leaks and ruptures are evaluated in Section 11.5 of this SER.

The new fuel storage area monitor is shown in DCD Tier 2, Figure 11.5-2O, "Location of Radiation Monitors at Plant (Auxiliary Building El. 156'-0")," as being located outside the new fuel area and on the other side of a wall. It was unclear to the staff as to why this location was appropriate for measuring the radiation levels in the new fuel storage area. Therefore, the staff issued **RAI 225-8254, Question 12.03-19 (ML15268A002),** requesting the applicant to justify the location of the new fuel storage area monitor.

In its **response to Question 12.03-19 (ML15365A240)**, the applicant indicated that the new fuel storage area monitor location shown in DCD Tier 2, Figure 11.5-2O, was the location of the monitor electronics and display. The applicant's response proposed revising DCD Figure 11.5-2O, to show the actual location of the radiation detector, located on the wall next to and above the new fuel storage area. This location is acceptable, however, as discussed below (in the discussion regarding RAI 225-8254, Question 12.03-15), it was unclear to the staff, if the locations of other monitors shown in the DCD 11.5-2 figures are the locations of the electronics or the detector itself. Therefore, the **staff closed RAI 225-8254, Question 12.03-19, as an unresolved item** and issued **RAI 376-8496, Question 12.03-49 (ML16028A036)**, as a follow-up to several RAIs regarding radiation monitoring. Question 12.03-49, is discussed below, following the discussion of RAI 225-8254, Question 12.03-15.

Similar to the new fuel storage area monitor, the main steam line monitors were also shown in the DCD figures as being located in different rooms, on the opposite side of the walls of the main steam lines. As a result, the staff issued **RAI 225-8254, Question 12.03-15 (ML15268A002)**, requesting that the applicant provide additional information regarding why the main steam line monitors (RE-217, RE-218, RE-219, and RE-220) were located on the other side of a wall from the main steam lines in DCD Tier 2, Figure 11.5-2K, "Location of Radiation Monitors at Plant (Auxiliary Building El. 137'-6")," and Figure 11.5-2L, "Location of Radiation Monitors at Plant (Auxiliary Building El. 137'-6")." In addition, in Question 12.03-15, the staff asked for additional informational information regarding the design, location, and function of monitors associated with the main steam lines.

In its **response to Question 12.03-15 (ML15365A240)**, the applicant indicated that the locations of the monitors shown in DCD Tier 2, Figure 11.5-2K and Figure 11.5-2L, were not the locations of the radiation monitor (detector), but were instead the locations of the electronics and display units for the monitors. Therefore, the applicant modified Figures 11.5.2K and 11.5-2L, to show the locations of the radiation detectors, which are located on the main steam lines. In addition, the applicant modified the locations of area monitors RE-237 and RE-238 (main steam and feedwater containment piping penetration area monitors) in the same figures. These monitors are now located behind a concrete column and, are no longer in a direct streaming path from the main steam piping penetrations. The proposed changes are acceptable, as the locations are in accordance with the SRP and ANSI/ANS-HPSSC-6.8.1-1981. However, in its response to RAI 225-8254, Question 12.03-19 (as discussed above), the applicant had indicated that the location of the new fuel storage area monitor shown in DCD Tier 2, Figure 11.5-2O, was also not the location of the detector, but is instead the location of the monitor and display. Therefore, it is unclear to the staff, if the locations of other monitors in the DCD 11.5-2 figures, represent the location of the radiation detector or the location of the electronics and display units. Without knowing this information, the staff is unable evaluate the placement of radiation monitors in accordance with the SRP and to determine if the criteria of ANSI/ANS-HPSSC-6.8.1-1981, is being met.

In order to resolve these issues, the **staff closed RAI 225-8254, Question 12.03-15, as an unresolved item** and issued **RAI 376-8496, Question 12.03-49 ML16028A036),** requesting that the applicant provide additional information regarding whether if the locations of all other monitors in the DCD 11.5-2 figures are the locations of the radiation detectors or the electronics and displays and to ensure that the actual locations are appropriately identified. In Question 12.03-49, the staff also requested that the applicant ensure that the detector locations were appropriate for the monitor function and to ensure that display panels and electronics are located in areas that would not result in operators accessing the display panels and electronics receiving high radiation levels.

In its **response to Question 12.03-49 (ML16123A127)**, the applicant indicated that it reviewed the area radiation monitors listed in DCD Tier 2, Table 12.3-6, to correct discrepancies and omissions. The applicant proposed revisions to DCD Tier 2, Figures 11.5-2A through Figure 11.5-2BB, to show the correct location of the radiation detectors and electronics/displays. The applicant also proposed updates to DCD Tier 2, Subsection 12.3.4.1.3, “General System Description,” to include information on the radiation monitor design and the locations of the monitors, the electronics/displays, and alarms. The electronics/displays are provided in separate locations for some monitors to alert workers of unfavorable radiological conditions before entering an area. This approach helps ensure that workers aren’t unnecessarily exposed to high radiation levels, and is acceptable. The staff reviewed the locations of the area radiation monitors and electronics/displays and alarms and generally found them to be consistent with the criteria of ANSI/ANS-HPSSC-6.8.1-1981 and to be acceptable. However, for a few areas the design did not meet the criteria in ANSI/ANS-HPSSC-6.8.1-1981. For example, the hot machine shop room had a radiation monitor but no alarm inside the room. Instead the alarm was located in the access area outside the room. The applicant indicated that the workers would be aware of a spill or event that would result in high radiation levels inside the room and know to evacuate the area. Therefore, no alarm was needed inside the room. This is inconsistent with ANSI/ANS-HPSSC-6.8.1-1981, which indicates that both audible and visual alarms should be provided inside the room with the monitor, and is unacceptable. The applicant also did not include any kind of alarm outside of the instrument calibrator facility area, which is a VHRA during calibration activities. This is also inconsistent with the guidance of ANSI/ANS-

HPSSC-6.8.1-1981. In addition, 10 CFR 20.1602 and RG 8.38, indicate that additional controls beyond those of normal high radiation areas, should be provided for VHRAs. Therefore, although the area is only a VHRA in the direct beam of the calibrator, additional controls should be provided beyond a locked door and a monitor inside the room, consistent with this guidance and requirement.

In **Revision 1 of its response to Question 12.03-49 (ML16211A098)**, the applicant updated the DCD to include an audible and visual alarm inside the hot machine shop room. However, the staff noted that while there are radiation detectors inside each of the truck bays and inside the waste drum storage area, there were not any audible or visual alarms provided inside those areas either. Instead the alarms (as well as electronics/displays) are located outside of the area and down a hallway (see DCD Tier 2, revised Figure 11.5-2T,"Location of Radiation Monitors at Plant (Compound Building El. 100'-0")," provided in the response). As indicated above, this is inconsistent with ANSI/ANS-HPSSC-6.8.1-1981. Therefore, the applicant should revise its response to RAI 376-8496, Question 12.03-49, to provide additional information. Hence, **RAI 376-8496, Question 12.03-49, is being tracked as an Open Item**.

In Revision 1 of its response to RAI 376-8496, Question 12.03-49, the applicant indicated that no alarm was included outside of the entrance to the instrument calibrator facility. The applicant also proposed updating DCD Tier 2, Subsection 12.3.4.1.5, "ARMS" (area radiation monitoring system), to provide information on the access requirements to the instrument calibrator facility area. The new information was essentially the same as the information provided in the original response and indicated that the room is provided with a locked door with a latch bolt which can only be opened from the outside by key.

Upon reviewing 10 CFR Part 36, "Licenses and Radiation Safety Requirements for Irradiators,", the staff determined that the instrument calibrator met the definition of an irradiator in 10 CFR 36.2, "Definitions," which defines an irradiator as, "a facility that uses radioactive sealed sources for the irradiation of objects or materials and in which radiation dose rates exceeding 5 grays (500 rads) per hour exist at 1 meter from the sealed radioactive sources in air or water, as applicable for the irradiator type, but does not include irradiators in which both the sealed source and the area subject to irradiation are contained within a device and are not accessible to personnel." Since the instrument calibrator, as described in the APR1400 application, meets this definition, it is subject to the requirements of 10 CFR Part 36, if incorporated into the APR1400 DC application. The current design does not address or meet several of these requirements, for example, 10 CFR 36.23(c) requires in part, that the radiation monitor must be integrated with personnel access door locks to prevent room access when radiation levels are high and attempted personnel entry while the monitor measures high radiation levels, must activate an alarm which alerts another individual knowledgeable of how to handle the situation of the attempted entry.

Since the APR1400 instrument calibrator facility does not appear to address several of the requirements of 10 CFR Part 36, the staff will issue an RAI requesting the applicant to demonstrate compliance with the requirements of 10 CFR Part 36. **This is an Open Item and is being tracked under RAI 376-8496, Question 12.03-49**, until the follow-up RAI is issued.

In **RAI 225-8254, Question 12.03-16 (ML15268A002),** the staff requested that the applicant provide clarification and address apparent discrepancies between monitor information, such as the radiation detection range of the detectors, in DCD Tier 2, Table 7.5-1, "Accident Monitoring Instrumentation Variables," Table 11.5-1, and Table 12.3-6.

In its **response to Question 12.03-16 (ML15365A240)**, the applicant proposed removing the detector range information specified in DCD Tier 2, Table 7.5-1, and simply referencing Tables 11.5-1 and 12.3-6, for the monitor ranges. This eliminated discrepancies and possible confusion regarding the ranges and other information associated with the monitors. The staff determined that this approach is acceptable. Therefore, **RAI 225-8254, Question 12.03-16, is being tracked as a confirmatory item** until the proposed changes have been incorporated into the DCD.

The containment upper operating area radiation monitors (high range) are required for monitoring radiation levels inside of containment during an accident in accordance with 10 CFR 50.34(f)(2)(xvii). However, TS Table 3.3.11-1, "Accident Monitoring Instrumentation," lists these monitors as having required action "F" if both channels failed, meaning that if both monitor channels failed, and the monitors were not restored to operable status within seven days, the applicant would simply be required to issue a report, and not be required to fix the monitor channels within any specified timeframe. For most other accident monitoring instrumentation, if both channels fail in that table, the plant is required to shut down. Since the monitors are required for monitoring containment during post-accident conditions and no alternative monitoring capabilities or procedures were described in the application, the staff issued **RAI 235-8275, Question 12.03-40 (ML15296A006)**, requesting that the applicant re-designate the containment upper operating area monitors as designation "E" requiring that if the monitors cannot be restored to operating status within the allotted timeframe the plant be required to shut down until the monitors are appropriately restored to service, as specified in the TS. This is consistent with the requirements of 10 CFR 50.34(f)(2)(xvii), which states that there must be a capability to monitor containment radiation levels during accident conditions. In its **response to RAI 235-8275, Question 12.03-40, Revision 1 (ML16142A035)**, the applicant proposed modifying the designation of these monitors to "E." Therefore, **RAI 235-8275, Question 12.03-40, is being tracked as a confirmatory item**, pending the incorporation of the proposed DCD changes.

In addition to the area radiation monitors provided in DCD Tier 2, Table 12.3-6, the MCR air intake monitors (RE-071A, RE-072B, RE-073A, and RE-074B), provided in DCD Tier 2, Table 11.5-1, are also relevant to the protection of plant workers. These monitors are safety-related beta monitors, which monitor gaseous radioactivity in the air intakes to the MCR. Upon detection of high radiation levels in the MCR intakes, these monitors initiate emergency filtration to ensure the dose to the operators remains below the 5-rem (50 mSv) limit, required by GDC 19. The acceptability of the design of these monitors is discussed in more detail in Section 14.3.8 of this SER (see the discussion related to RAI 116-8054, Question 14.03.08-5).

While there are process and effluent monitors, as discussed in DCD Tier 2, Section 11.5, there are no fixed airborne radiation monitors provided in the plant design for the purpose of detecting airborne activity levels inside the plant. However, portable airborne activity monitors will be provided, as specified in NEI 07-03A, which is referenced in DCD Tier 2, Section 12.1. As specified in DCD Tier 2, Section 12.3.3, these monitors will be used in areas that are normally occupied and have a significant potential for airborne contamination and can detect the time-integrated change of the airborne radioactivity within 10 DAC-hours for the most limiting particulate and iodine species in each area. A monitor that is able to detect the time-integrated change of the airborne radioactivity within 10 DAC-hours is consistent with the guidance in SRP Section 12.3-12.4 and is acceptable. Because it is impossible to know where leaks or spills will result in high levels of radioactivity in the design stage, it is appropriate to set up portable airborne activity monitors, as needed during plant operation.

In addition, COL Information Item 12.3(1) indicates that the COL applicant is to provide portable instruments and the associated training and procedures in accordance with 10 CFR 50.34(f)(2)(xxvii) and the criteria in Item III.D.3.3 of NUREG-0737. However, this COL information item does not specify whether the portable monitors will meet any other guidance, beyond the guidance for post-accident conditions. Therefore, the staff issued **RAI 235-8275, Question 12.03-32 (ML15296A006)**, requesting that the applicant provide this information.

In its **response to Question 12.03-32 (ML16036A042)**, the applicant proposed updating COL Information Item 12.3(1) to specify that the COL applicant, in addition to providing portable instruments in accordance with 10 CFR 50.34(f)(2)(xxvii) and NUREG-0737, will also provide portable instruments and associated training and procedures in accordance with RG 8.8. This ensures that the applicant also provides portable instruments and training and procedures for portable instruments during normal operation. Therefore, the response is acceptable and **RAI 235-8275, Question 12.03-32, is being tracked as a confirmatory item**, pending the incorporation of the proposed DCD changes.

COL Information Item 12.3(2) indicates that the COL applicant is to determine the WARN and ALARM setpoints of the area radiation monitors based on the site-specific conditions and operational requirements. Some of the area radiation monitors provide automatic safety functions, as discussed above. However, this COL information item does not specify if the COL applicant will provide the setpoints for the safety functions. Therefore, the staff issued **RAI 235-8275, Question 12.03-33 (ML15296A006),** requesting that the COL applicant provide this information.

In its **response to Question 12.03-33 (ML16036A042)**, the applicant proposed updating COL Information Item 12.3(2) to state that the COL applicant will also provide the containment purge isolation and fuel handling area emergency ventilation actuation signals. This ensures that the setpoints for the safety-functions will also be specified by the COL applicant. In addition, DCD Tier 2, Subsection 12.3.4.1.6, “Range and Alarm Setpoints,” which is referenced in Chapter 11 for determining the setpoints for safety-related monitors, already states that plant procedures will determine the setpoints for safety-related components and states that they must be in conformance with American National Standard Institute/International Society of Automation (ANSI/ISA)-67.04-1994, “Setpoints For Nuclear Safety-Related Instrumentation,” which is referenced in RG 1.105, “Setpoints for Safety-Related Instrumentation.” DCD Tier 2, Chapter 7, “Instrumentation and Controls,” also provides information on the setpoint methodology. So, the DCD already provides information indicating that the setpoints for the safety-related monitors (including those in Chapter 11) will be determined appropriately by plant procedures. Therefore, **RAI 235-8275, Question 12.03-33, is being tracked as a confirmatory item**, pending the incorporation of the proposed DCD changes.

Other aspects of radiation monitor design, function, and placement, including ITAAC associated with radiation monitoring are discussed in Section 14.3.8 of this SER (see the discussions related to the response to RAI 116-8054, Question 14.03.08-4 and Question 14.03.08-5 and RAI 368-8470, Question 14.03.08-14).

Upon complete resolution of the open items above and in Chapter 14, “Verification Programs,” of this SER, the staff will determine if the area radiation and airborne radioactivity monitoring instrumentation of the APR1400 design meets the applicable NRC requirements.

12.3.4.1.5 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 the environment, as well as the generation of radioactive waste. Applicants are also required to describe how it will facilitate decommissioning. The guidance in SRP Section 12.3-12.4, 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 contains a basis acceptable to the staff for complying with the requirements of 10 CFR 20.1406. Wherever the applicant adhered to this guidance, the staff can have reasonable assurance of compliance with 10 CFR 20.1406.

DCD Tier 2, Section 12.4.2, "Minimization of Contamination and Radioactive Waste Generation," 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. These features include measures to minimize facility contamination and contamination of the environment, and features to facilitate decommissioning. DCD Tier 2, Table 12.4-10, "NRC RG 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste," provides a listing of many of the specific features in the APR1400 design consistent with the guidance of RG 4.21, and the requirements of 10 CFR 20.1406. In addition, the DCD references NEI 08-08A "Generic FSAR Template Guidance for Life Cycle Minimization of Contamination" and includes COL Information Item 11.2(9), which specifies that the COL applicant is to develop a plant-wide RG 4.21 program following the guidance of NEI 08-08A for contamination control. This ensures that the program will meet the requirements of 10 CFR 20.1406 for life-cycle minimization of contamination.

The design includes design features to prevent and minimize contamination, such as segregation of components, flushing capabilities for components and piping that handle or transport radioactive fluid (such as resin transfer lines and other components which transport or contain radioactive fluids) to prevent crud buildup and hot spots, the use of containment walls or dikes around potentially contaminated structures, systems, and components (SSCs), minimization of buried and embedded piping and drains, and use of corrosion resistant materials. Early leak detection capability is built into each cubicle that may contain radioactive or potentially radioactive fluid for quick assessment, as discussed in DCD Tier 2, Subsection 12.4.2.4.1, "Early Leak Detection Drain Pipe." Leaks are identified by alarms in the radwaste control room and MCR. A leak collection trench is provided within the basemat of the auxiliary building and compound building which flows to a sump. The trench is epoxy coated to facilitate drainage and prevent leakage through the trench. Leak detection equipment is located as close as practicable to potential leaks to provide for prompt and accurate leakage detection.

Piping that extends from one building to another with a gap in between buildings is equipped with piping sleeves or double-walled concentric piping to prevent direct, unintended leakage to the environment. Embedded and buried piping is minimized to the extent practicable. The application also indicated that when embedded piping cannot be avoided, double-walled piping is used and the outside pipe is designed to drain to a local sump or onto the floor for drainage into a nearby sump. The staff issued **RAI 235-8275, Question 12.03-42 (ML15296A006),** requesting that the applicant justify why allowing the leakage to drain onto the floor is commensurate with 10 CFR 20.1406.

In its **response to Question 12.03-42 (ML16057A071)**, the applicant proposed removing the sentences from DCD Tier 2, Subsections 12.4.2.4.3, “Penetration Design between Buildings,” and 12.4.2.4.4, “Minimization of Embedded and/or Buried Piping,” which indicated that radioactive liquid would drain onto the floor before going to a nearby sump, and revised to indicate that leakage is routed directly to the nearby sump. The applicant also provided additional information on the piping design and how leaks will be detected, especially with regards to embedded piping. The applicant’s response indicated that although buried piping is minimized in the APR1400 design (as specified in DCD Tier 2, Subsection 12.4.2.1, Design Considerations”), embedded piping above the basemat level of the plant buildings and carrying potentially contaminated radioactive fluid is to be routed in double-walled piping. The leakage from the inner piping is collected in the outer-walled pipe. When the piping emerges from the wall or floor, the leakage is drained to a nearby collection point, such as a drain collection drain pipe equipped with a water trap and a level switch. When liquid is accumulated to a predetermined level, the level switch initiates an alarm in the MCR for operator actions. For embedded piping at the basemat level, the leak detection piping (e.g., from the liquid radwaste system floor drain tanks, equipment waste tanks, etc.) is routed from the cubicle to the nearby drain header or sump. This category of leak detection piping is generally short in order to facilitate the maximum slope for leakage collection for early detection, which will facilitate timely operator actions to assess the situation and perform any remediation. The leak detection piping is provided with double-wall piping, with any leakage from the inner pipe draining into the concrete trenches. This design is consistent with the guidance of RG 4.21 and the requirements of 10 CFR 20.1406, and is therefore, acceptable. As a result, RAI **235-8275, Question 12.03-42, is being tracked as a confirmatory item**, pending the incorporation of the proposed DCD revisions.

In addition, the APR1400 design includes a separate CCW structure, which house the CCW heat exchangers. The CCW stricture is located near the service water system, and includes a sump to collect leakage from the CCW heat exchangers. However, it is unclear where the collected liquid in the sump is routed to (for example, to the LWMS) and if the sump included design features to prevent the release of radioactive material (such as sealer for the sump and seals around piping). Therefore, the staff issued **RAI 225-8254, Question 12.03-11 (ML15268A002)**, requesting that the applicant provide this information.

In its **response to Question 12.03-11 (ML16050A007)**, the applicant indicated that if the radiation monitors detect radiation levels from the CCW structure sump liquids above a pre-determined setpoint, the collected liquid is routed to the LWMS for treatment. The applicant also proposed updating DCD Tier 2, Subsection 9.2.2.2.5, “Design Features for Minimization of Contamination,” “Prevention/Minimization of Unintended Contamination,” Item c to specify this information and to provide additional 10 CFR 20.1406, related design features, including to specify that the CCW cubicles are designed with sloped floors and are epoxy coated.

While this information was acceptable, it contained numerous apparent errors and inconsistencies. For example, while the response indicated that there are radiation monitors provided in the CCW sump, there were no such radiation monitors provided in DCD Tier 2, Chapter 11.

In **Revision 1 of its response to Question 12.03-11 (ML16211A139)**, the applicant revised the approach indicating that the radioactive material from the CCW sump is routed either to the LWMS or to the turbine generator building sump where it will then be sent for treatment and release. The staff finds that this approach is acceptable because the release of radioactive material is still being controlled and measured. The applicant also specified that new COL

Information Item 9.3(5) will be included in its response to RAI 244-8326, Question 09.03.03-4 (ML16210A149), to state that the COL applicant is to provide the flow diagram of the CCW heat exchanger building drain system to the LWMS or turbine generator building sump, as well as the flow diagram for the turbine building drain system and the interconnection from the auxiliary boiler building sump. The staff confirmed that the revised response to RAI 244-8326, Question 09.03.03-4, included the proposed COL Information Item 9.3(5), which is acceptable. The revised response to RAI 244-8326, Question 09.03.03-4, also provided additional information related to the 10 CFR 20.1406 design features and clarified the location of the condensate receiver tank radiation monitor (RE-103). The staff will confirm that these changes are included into the DCD. Finally, the applicant has indicated that its response to RAI 132-8088, Question 11.05-2, would be updated to Revision 2 to include the CCW sump monitors and missing turbine building sump monitors in DCD Tier 2, Section 11.5. The second revised response to RAI 132-8088, Question 11.05-2, has yet to be submitted. Therefore, the applicant's response to **RAI 225-8254, Question 12.03-11, is being tracked as an Open Item**, until the applicant updates its response to Question 11.05-2, to incorporate the radiation monitors described in the response to Question 12.03-11, into the DCD.

The DCD did not discuss where leakage that is collected in the holdup tank, RMWTs, and BAST sumps, located outside in a tank house, would be routed. The staff issued **RAI 225-8254, Question 12.03-12 (ML15268A002),** requesting that the applicant provide this information.

In its **response to Question 12.03-12 (ML16027A034)**, the applicant updated the DCD to specify that in the event leakage is detected in the sump of the tank house of the outdoor tanks, a level switch sends a signal to the MCR for operator action. At which point, the fluid that is collected in the sump is routed to the auxiliary building equipment drain sump, where it can be transferred to the LWMS. The staff determined that this design feature is consistent with the guidance of RG 4.21 and the requirements of 10 CFR 20.1406, and is therefore, acceptable. As a result, **RAI 225-8254, Question 12.03-12, is being tracked as a confirmatory item**, pending the incorporation of the proposed DCD changes.

The application also specified that sumps in the reactor containment building are small and that frequent transfers of material to the LWMS will occur. Therefore, the staff issued **RAI 235-8275, Question 12.03-36 (ML15296A006),** requesting that the applicant specify if the sumps are large enough to allow for access for cleaning and maintenance of the sump pumps.

In its **response to Question 12.03-36 (ML15334A454)**, the applicant indicated that there are two sumps in the containment building (the reactor containment building drain sump and the ICI cavity sump). Both sumps are designed to facilitate repair of their pumps by taking into account pump maintenance and removal, including equipment removal pathway, personnel accessibility, areas for maintenance and removal activities, and provisions for the removal of the sump pumps. In addition, the applicant specified that the sumps are designed so that work on the sump pumps can be performed without a need to set up any platform and that the sump areas are coated in epoxy for the ease of area decontamination. These and other design features will ensure that contamination is minimized and that doses will be ALARA for any work that is required on the sump pumps. The staff finds, while the applicant provided the above information in the RAI response, it did not propose including any of the additional information in the DCD. Therefore, **RAI 253-8275, Question 12.03-36, is unresolved/closed**. The staff issued **RAI 396-8463, Question 12.03-51 (ML16034A054),** requesting that the applicant provide additional information in the DCD regarding how the sumps and sump pumps are designed to minimize contamination and to ensure that worker exposure will be ALARA.

In its **response to Question 12.03-51 (ML16063A116)**, the applicant proposed adding the requested information to DCD Tier 2, Subsection 9.3.3.2.4.d, “Reactor containment building equipment and floor drains.” This information provides design features consistent with the requirements of 10 CFR 20.1101(b) and 10 CFR 20.1406. Therefore, the staff determined that the **response to RAI 396-8463, Question 12.03-51, is acceptable, pending the proposed DCD revisions. This question is being tracked as a confirmatory item**, pending the incorporation of the proposed DCD updates.

In reviewing the CVCS figures provided in the DCD Tier 2, Figure 9.3.4-1, the staff identified a piping connection between the RMWT and the fire protection system with only one diaphragm valve separating the potentially contaminated RMWT from the fire protection system. The applicant indicated in a teleconference that this line was only used for the emergency containment spray backup system during accident conditions, and that there are no high pressures associated with the valve. For these reasons, the applicant indicated that the valve is not expected to be operated frequently or to leak. However, DCD Tier 2, Subsection 9.1.3.2.3, “Design Features for Minimization of Contamination,” states that, “utility connections are designed with a minimum of two barriers to prevent the contamination of clean systems,” and RG 4.21, Section A-1, paragraph “a” states that, “The interface between structures, systems, and components (SSC) important to radiological safety and nonradioactive SSCs should be minimized. Necessary interfaces should have a minimum of two barriers, including one that can be a pressure differential, and should have instrumentation for prompt detection and control of cross-contamination.” In addition, operating experience indicates that diaphragm valve have a tendency for through leakage, especially as the valves age. Regarding minimizing the potential of cross contaminating the fire protection system, the staff issued **RAI 504-8628, Question 12.03-54 (ML16193A628)**, requesting additional information on design of the connection between the RMWT and the fire protection system, consistent with DCD Tier 2, Subsection 9.1.3.2.3 and RG 4.21.

In its **response to Question 12.03-54 (ML16265A641)**, the applicant proposed revising DCD Tier 2, Figure 9.3.4-1, replacing the diaphragm valve with two gate valves and connections for a flexible hose which can be connected between them. This design is consistent with RG 4.21 and 10 CFR 20.1406 and is therefore, acceptable. **The response is being tracked as a confirmatory item**, pending the incorporation of the proposed DCD change.

In addition, DCD Tier 2, Subsection 12.4.2.4.4, indicates that the COL applicant will implement concrete tunnels for those systems that include underground piping carrying contaminated or potentially contaminated fluid, see COL Information Item 12.4(3) below. While the DCD indicates that the tunnels will be coated with epoxy coating and include liquid detection level switches and that liquid detection will be detected in the MCR, it is unclear if the liquid detection capability will be capable to detect minor leaks, as specified in RG 4.21. Therefore, the staff issued **RAI 14-7858, Question 12.03-1 (ML15142A608)**, requesting that the applicant provide additional clarification on this issue.

In its **response to RAI 14-7858, Question 12.03-1 (ML15201A377)**, the applicant proposed new COL Information Item 12.4(3) in a future DCD update. The addition specified that the tunnels will be coated with epoxy and equipped with sumps with liquid detection level switches that will alarm in the MCR. Since the tunnels are coated with epoxy and routed to a sump, relatively minor leaks should be detected by accumulation of fluid in the sump. The staff determined that this design is consistent with RG 4.21 and is, therefore, acceptable. Hence, **RAI 14-7858, Question 12.03-1, is being tracked as a confirmatory item**, until the proposed changes have been incorporated into the DCD.

In addition, COL Information Item 12.4(2) indicates that the COL applicant is to provide operational procedures and programs, including the development of a site radiological environmental monitoring program, to implement the minimization of contamination approach. Since the COL applicant is responsible for site specific design and operating procedures, it is acceptable for the COL applicant to provide this information. However, this COL item does not specify if the COL applicant will provide information on how site-specific design features will meet the requirements of 10 CFR 20.1406 and the guidance of RG 4.21, including whether mobile radwaste processing equipment will meet these requirements. The staff issued **RAI 235-8275, Question 12.03-37** and **Question 12.03-39 (ML15296A006)**, requesting that the applicant provide this additional information.

In its **response to Question 12.03-37 (ML16028A286)**, the applicant proposed updating COL Information Item 11.2(6) and COL Information Item 11.4(3), to specify that any mobile and temporary equipment will follow the guidance of NUREG-0800 and RG 4.21. In addition, in its **response to Question 12.03-39 (ML16057A071)**, the applicant proposed updating COL Information Item 12.4(2), to clarify that operational procedures and programs for minimizing contamination will be in accordance with 10 CFR 20.1406 and RG 4.21. Furthermore, in its **response to RAI 14-7858, Question 12.03-3 (ML15201A377)**, the applicant also proposed updating COL Information Item 12.4(2), to specify that the COL applicant will also specify that the 10 CFR 20.1406 program will also be in accordance with the applicable portions of RG 4.22, "Decommissioning Planning During Operation," and the documentation required by 10 CFR 20.1501, "General." These responses clarify that the COL applicant should provide mobile and temporary equipment, along with procedures and programs in accordance with the applicable requirements and guidance. Therefore, the applicant's responses are acceptable and **RAI, Questions 12.03-3, 12.03-37, and 12.03-39, are being tracked as confirmatory items**, pending the incorporation of the proposed DCD revisions.

Upon complete resolution of the open items above, the staff will determine if the measures to minimize contamination for the APR1400 design meet the applicable NRC requirements.

12.3.4.2 Dose Assessment

The staff reviewed DCD Tier 2, Section 12.4.1, “Dose Assessment,” for completeness against the guidelines in RG 1.206 and the criteria set forth in SRP Section 12.3-12.4. The staff ensured that the applicant had either committed to follow the guidance of the applicable RGs and staff positions set forth in NUREG-0800, Section 12.3-12.4, or provided acceptable alternatives. Wherever the DCD 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 PWRs. Radiation exposures to operating personnel shall not exceed the occupational dose limits specified in 10 CFR 20.1201, and doses should be ALARA in accordance with 10 CFR 20.1101.

In DCD Tier 2, Section 12.4.1, the applicant provided an assessment of the annual occupational radiation dose that would be received by the operating staff of an APR1400 facility. DCD Tier 2, Tables 12.4-1 through 12.4-7, 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 0.585 person-Sv (58.5 person-rem). The DCD does not contain a separate determination of doses attributable to airborne activity; however, experience at operating light water reactors (LWRs) demonstrates that the doses from airborne radioactivity is not a significant contribution to the total dose. Because the expected airborne doses are not significant, it is acceptable that they are not included in this dose estimate.

The applicant used operating data from Hanul, Unit 3 (a Korean reactor which began operation in 1998) as a basis for determining dose estimates for the APR1400. It is consistent with the guidance of RG 8.19," Occupational Radiation Dose Assessment in Light-Water Reactor Power Plants -- Design Stage Man-Rem Estimates," to use operating data as part of the basis for the dose assessment. The cumulative annual dose of 0.585 person-Sv (58.5 person-rem) for operating a plant is consistent with the Electric Power Research Institute URD design guideline of 1.0 person-Sv (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-Sv [69 person-rem]). In addition, since the APR1400 design includes additional design features to limit corrosion products, the applicant indicates that doses should be less than those estimated.

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 the high dose maintenance activities on plugged piping and pumps. These design features conform to the guidance of RG 8.8 and are, therefore, acceptable.

During construction of the plant, construction workers may be exposed to various radiation sources due to site-specific factors including if there are existing nuclear units on site, other units being constructed at the same time, and radiography sources. In RG 1.206, Section C.I.12.3.5, “Dose Assessment,” the guidance is that all applicants with multi-unit sites should provide estimated annual doses to construction workers due to onsite radiation sources from existing operating plant(s). Since this information is site-specific, the applicant included COL Information Item 12.4(1) requesting the COL applicant to provide this information. This approach is acceptable. However, the COL information item was specific regarding what the COL applicant’s analysis would encompass and it did not specify all potential considerations for calculating construction worker dose. In order to resolve this issue, the staff issued **RAI 14-7858, Question 12.03-2 (ML15142A608),** requesting the applicant provide information related to the COL item to ensure the COL applicant considers all factors that could contribute to construction worker dose.

In its **response to RAI 14-7858, Question 12.03-2 (ML15201A377)**, the applicant proposed to reword COL Information Item 12.4(1), to clarify that the COL applicant will consider all factors that could contribute to dose to construction workers, when estimating construction worker dose. The staff finds that the revised COL item is consistent with the guidance of RG 1.206 and is, therefore, acceptable. Hence, **RAI 14-7858, Question 12.03-2, is being tracked as a confirmatory item**, until the proposed changes have been incorporated into the DCD.

12.3.5 COL Information Items

The following is a list of items from Table 1.8-2 of the DCD, as they are provided in the initially accepted application (the applicant has proposed revising some of these COL items, as discussed above):

- COL 12.3(1): The COL applicant is to provide portable instruments and the associated training and procedures in accordance with 10 CFR 50.34(f)(2)(xxvii) and the criteria in Item III.D.3.3 of NUREG-0737.

- COL 12.3(2): The COL applicant is to determine the WARN and ALARM setpoints of the ARMS based on the site-specific conditions and operational requirements.

- COL 12.4(1): The COL applicant is to estimate construction worker doses based on site-specific number of operating units, distances, meteorological conditions, and construction schedule.

- COL 12.4(2): The COL applicant is to provide operational procedures and programs, including the development of a site radiological environmental monitoring program, to implement the minimization of contamination approach.

- COL 12.4(3): The COL applicant is to implement concrete tunnels for piping of the systems that may include underground piping carrying contaminated or potentially contaminated fluid to minimize buried piping.

12.3.6 Conclusion

Upon the resolution of the **Open Items tracked under Questions 12.03-8, 12.03-10, 12.03-11, 12.03-13, 12.03-26, 12.03-43, 12.03-46, 12.03-49,** and **12.03-53,** the staff will determine if the APR1400 facility design features meet the relevant requirements of 10 CFR Part 20, 10 CFR Part 50, and 10 CFR Part 52, as they relate to this section.

12.4 Dose Assessment and Minimization of Contamination

The staff's review of this section of the DCD is documented under Subsection 12.3.4.2 of this SER.

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 ORE will be ALARA, must be in place. This includes procedures used in normal operation, refueling, in-service inspections, handling of radioactive material, spent fuel handling, routine maintenance, and sampling and calibration related to radiation safety.

12.5.2 Summary of Application

DCD Tier 1: There are no DCD Tier 1 entries for this area of review.

DCD Tier 2: The applicant has provided a DCD Tier 2 system description in Section 12.5, summarized here in part, as follows:

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

12.5.3 Regulatory Basis

The relevant requirements of NRC regulations for the operational radiation protection program, and the associated acceptance criteria, are in NUREG-0800, Section 12.5, “Operational Radiation Protection Program.” The guidelines in SRP section 12.5 and the applicable regulatory requirements will be addressed by the staff during the review of COL applications.

12.5.4 Technical Evaluation

The APR1400 DCD Tier 2 Section 12.5, states that the COL applicant is to provide the radiation protection program. The DCD also states that the COL applicant is to provide the operational radiation protection program for ensuring that OREs are ALARA. It adds that the programs include the radiation protection program, ALARA program, groundwater protection program, and leakage control programs; and will include some of the following aspects:

Organizational positions, functional responsibilities, experience, and qualifications of personnel.

Equipment necessary to measure radiation including the number, type, range, and sensitivities of equipment necessary, as well as the calibration methods and frequencies; and the planned use of portable, fixed, laboratory, and personnel monitoring instruments.

Appropriate health physics facilities and associated protective equipment for controlling ORE and contamination (while the APR1400 design does include areas for health physics operations, decontamination facilities, and storage of health physics equipment, it is up to the COL applicant to determine how these areas will be utilized and if additional facilities are necessary).

- Methods for ensuring development of the training, retraining, and indoctrination program and the radiation protection instruction manuals.

- Procedures to receive, store, transfer, and dispose of radioactive material, minimize contamination, and facilitate decommissioning.

In addition, as discussed in Section 12.1 of this SER, the DCD states that the COL applicant will develop the organizational structure for the radiation protection program and will satisfy applicable regulations and RGs 1.33, 1.8, 8.8, and 8.10, “Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as Is Reasonably Achievable.” Also discussed in Section 12.1 above, the COL applicant is to describe how the plant follows the guidance provided in RGs 8.2, 8.7, 8.9, 8.13, 8.15, 8.20, 8.25, 8.26, 8.27, 8.28, 8.29, 8.34, 8.35, 8.36, and 8.38.

In addition, SRP Section 12.5, specifies that the staff will review information describing the implementation of RG 8.4. In **RAI 6-7854, Question 12.05-1 (ML15155B329)**, the staff requested that the applicant specify how the guidance of RG 8.4 will be met or provide an alternative approach. In the response to **RAI 6-7854, Question 12.05-1, (ML15166A246)**, the applicant indicated that COL Information Item 12.1(3) will be updated in a future DCD revision to indicate that the COL applicant will also describe how the plant follows the guidance in RG 8.4. The staff determined that this approach acceptable. Therefore, **RAI 6-7854, Question 12.05-1, is being tracked as a confirmatory item**.

Therefore, the radiation protection program will be addressed by the COL applicant and the staff will conduct this review during the review of COL applications referencing the APR1400 DCD.

12.5.5 Combined License Information Items

The following is a list of items from Table 1.8-2, "Combined License Information Items," of the DCD:

- COL 12.5(1): The COL applicant is to provide the operational radiation protection program, including the items described in Section 12.5.

As discussed in the technical evaluation section, DCD Tier 2, Section 12.1 also contains COL information items relevant to the radiation protection program. Those items are discussed in Section 12.1 of this SER.

12.5.6 Conclusion

The staff determined that the deferral of the review of this area to the COL application review and the description of the information item are acceptable.