NUREG-0800

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

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12.3 ~~-~~– 12.4 RADIATION PROTECTION DESIGN FEATURES

REVIEW RESPONSIBILITIES

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

Secondary - None

I. AREAS OF REVIEW

The staff will review the applicant's ~~preliminary safety analysis report~~Preliminary Safety Analysis Report (PSAR) for a construction permit (CP) or ~~final safety analysis report~~Final Safety Analysis Report (FSAR) for an operating license (OL), design certification (DC), or combined license (COL~~),~~) Technical Submittal, as it relates to radiation protection design features, taking into account design dose rates, anticipated operational occurrences~~, and accident conditions~~ (AOO), and accident conditions. The staff will review radiation protection design features to ensure the applicant's design reflects Occupational Radiation Exposure (ORE) from direct and airborne radioactive material in the facility as low as is reasonably achievable (ALARA), control the exposure of members of the public to direct radiation from sources located at the facility, minimize contamination of the facility and the environment, minimize the generation of waste and protect equipment important to safety.

Draft Revision 5 – October 2012

USNRC STANDARD REVIEW PLAN

This Standard Review Plan (SRP), NUREG-0800, has been prepared to establish criteria that the U.S. Nuclear Regulatory Commission (NRC) staff responsible for the review of applications to construct and operate nuclear power plants intends to use in evaluating whether an applicant/licensee meets the NRC regulations. The SRP is not a substitute for the NRC regulations, and compliance with it is not required. However, an applicant is required to identify differences between the design features, analytical techniques, and procedural measures proposed for its facility and the SRP acceptance criteria and evaluate how the proposed alternatives to the SRP acceptance criteria provide an acceptable method of complying with the NRC regulations.

The standard review plan sections are numbered in accordance with corresponding sections in Regulatory Guide (RG) 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)." Not all sections of RG 1.70 have a corresponding review plan section. The SRP sections applicable to a combined license application for a new light-water reactor (LWR) are based on RG 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)."

These documents are made available to the public as part of the NRC's policy to inform the nuclear industry and the general public of regulatory procedures and policies. Individual sections of NUREG-0800 will be revised periodically, as appropriate, to accommodate comments and to reflect new information and experience. Comments may be submitted electronically by email to NRR_SRP@nrc.gov

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The specific areas of review are as follows:

1. Facility Design Features

A. In the CP PSAR, the DC FSAR~~,~~ or the COL FSAR, the description of equipment and facility design features used for assuring that ~~occupational radiation exposure (~~ORE~~)~~ will be ~~as low as is reasonably achievable (~~ALARA~~)~~, consistent with Title of the *Code of Regulations* (10 CFR) 20.1101.

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

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

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

E. Information describing design features that will facilitate eventual decommissioning and minimize, to the extent practicable, contamination of the facility and environment and the generation of radioactive waste in accordance with ~~Title 10 of the *Code of Federal Regulations* (10 CFR), Section~~10 CFR 20.1406 (CP PSAR and updates in the OL FSAR, DC FSAR, or the COL FSAR to the extent that they are not addressed in a referenced certified design~~)~~). The guidance contained in RG 4.21 and Appendix 12.3-12.4-A "Evaluation and Acceptance Criteria for 10 CFR 20.1406 to Support Design Certification and Combined License Applications," enclosed herein, is intended to be used by reviewers of this Standard Review Plan (SRP) section, and other SRP section reviewers, as applicable.

F. Information describing the implementation of RG 8.8 guidelines to reduce the production, distribution, and retention of activation products through the specification for materials and features of components that will be in direct contact with primary coolant, or provide information describing any proposed alternatives. (CP PSAR and updates in the OL FSAR, DC FSAR, or the COL FSAR to the extent that they are not addressed in a referenced certified design), consistent with the requirements of 10 CFR 20.1001(b) and 10 CFR 52.47(a)(22).

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G. Information describing the implementation of RG 8.8 guidelines on the specifications for equipment and components provided to improve reliability, reduce leakage, facilitate maintenance and reduce required inspections, as well as information describing any proposed alternatives. (CP PSAR and updates in the OL FSAR, DC FSAR, or the COL FSAR to the extent that they are not addressed in a referenced certified design), consistent with the requirements of 10 CFR 20.1001(b) and 10 CFR 52.47(a)(22).

H. Information describing the implementation of RG 8.8 guidelines on the specifications for station lighting features to provide a favorable working environment, promote work efficiency and facilitate egress from high radiation areas if the station lighting system fails, through the use adequate lighting, the use of extended service lamps, design features that permit the servicing of the lamps from lower radiation areas and provisions for emergency lighting, as well as information describing any proposed alternatives. (CP PSAR and updates in the OL FSAR, DC FSAR, or the COL FSAR to the extent that they are not addressed in a referenced certified design), consistent with the requirements of 10 CFR 20.1001(b) and 10 CFR 52.47(a)(22).

I. Information describing the implementation of RG 8.8 guidelines on the description and location of each very high radiation area on plant layout diagrams and the design features provided to control access to radiologically restricted areas (including potentially very high radiation areas), such as the reactor cavity and the fuel transfer tube during refueling operations, as well as information describing any proposed alternatives. (CP PSAR and updates in the OL FSAR, DC FSAR, or the COL FSAR to the extent that they are not addressed in a referenced certified design), consistent with the requirements of 10 CFR 20.1001(b) and 10 CFR 52.47(a)(22).

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2. Shielding

A. The shielding to be provided for each of the radiation sources identified in ~~safety analysis report~~Safety Analysis Report (SAR~~)~~), Chapter 11 and Section 12.2, and other applicable sections, including the design criteria and the shield material to be used for penetrations, to preclude radiation (including neutron) streaming into containment or other areas that may be occupied and for attenuation of neutrons streaming from the annulus between the reactor pressure vessel and biological shield (CP PSAR and updates in the OL FSAR, DC FSAR, or the COL FSAR to the extent that they are not addressed in a referenced certified design); specification of shield wall thicknesses for all shielded spaces on the plant layout drawings or in separate tables (as noted in ~~Item~~item I.1.C above~~)~~).

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

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

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

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

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

3. Ventilation

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

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

C. Information describing the application of RG 1.52 (particularly Sections C.3.10 and 4.10~~)~~)), RG 1.140 and RG 8.8, and information describing any proposed alternatives (CP PSAR and updates in the OL FSAR, DC FSAR, or the COL FSAR to the extent that they are not addressed in a referenced certified design~~)~~).

4. Area Radiation and Airborne Radioactivity Monitoring Instrumentation

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

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

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

D. Information describing the implementation of radiation monitoring equipment criteria listed in RGs~~ ~~ 8.2, 8.8, 8.25, ~~and Revision 4 of ~~RG 1.97, BTP 7-10 and American National Standards Institute/Health Physics Society (ANSI/HPS) Standard N13.1-1999, and information describing any proposed alternatives (CP PSAR and updates in the OL FSAR, DC FSAR, or the COL FSAR to the extent that they are not addressed in a referenced certified design~~)~~).

E. Description of the in~~-~~-containment high~~-~~-range radiation monitoring capability after an accident, in accordance with 10 CFR 50.34(f)(2)(xvii), ~~Item~~item II.F.1.3 of NUREG-0737, ~~and Revision 4 of ~~RG 1.97, BTP 7-10 and 10 CFR Part 50, Appendix E VI.2(a).

F. Description of in plant radiation airborne radioactivity monitoring system in accordance with 10 CFR 50.34(f)(2)(xxvii~~) and Item~~), item III.D.3.3 of NUREG-0737, RG 1.97 and BTP 7-10.

G. Description of locations for fixed radiation monitors in accordance with ANSI/American Nuclear Society (ANS~~)~~-)/Health Physics Society Standards Committee (HPSSC) Standard ANSI/ANS/HPSSC-6.8.1

H. Description of radiation monitors in areas where special nuclear material is handled or stored in accordance with 10 CFR 50.68 or 10 CFR 70.24.

5. Dose Assessment

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

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

C. For facilities being constructed adjacent to an existing operating nuclear unit(s), in accordance with the guidance contained in RG 1.206, Subsection C.I.12.3.5 provide a description of the basis for the dose assessment process for plant construction workers, providing detailed information as to the estimated number of construction workers and estimated annual doses (from direct, gaseous, and liquid sources) to these workers, in accordance with the provisions of NUREG-1555 (CP PSAR and updates in the OL FSAR, DC FSAR, or COL FSAR).

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6. Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC). For DC and COL reviews, the staff reviews the applicant's proposed ITAAC associated with the structures, systems, and components (SSCs) related to this ~~Standard Review Plan (~~SRP~~)~~ section in accordance with SRP Section 14.3, ~~“~~"Inspections, Tests, Analyses, and Acceptance Criteria~~.”~~." The staff recognizes that the review of ITAAC cannot be completed until after the rest of this portion of the application has been reviewed against acceptance criteria contained in this SRP section. Furthermore, the staff reviews the ITAAC to ensure that all SSCs in this area of review are identified and addressed as appropriate in accordance with SRP Section 14.3.

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

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

Review Interfaces

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Systems described in the Technical Submittal may differ from those outlined in the SRP. The staff should use the following recommended SRP section interfaces as the basis for reviewing other supplemental or complementary radiation protection design feature information provided in the Technical Submittal for a specific plant design:

3.8.3 CONCRETE AND STEEL INTERNAL STRUCTURES OF STEEL OR CONCRETE CONTAINMENTS – as it relates to the radiation protection design features provided to reduce ORE, protect plant equipment and controlling direct dose to members of the public; for areas that may contain irradiated fuel or irradiated components; for access points to Very High Radiation Areas; of structural shielding materials, including dimensions and specifications for materials used for shielding; to support inservice inspections of structures; provided to minimize contamination, to the extent practicable, of the facility or environment.

3.8.4 OTHER SEISMIC CATEGORY I STRUCTURES – as it relates to the radiation protection design features to reduce ORE and protect plant equipment; for areas that may contain irradiated fuel or irradiated components; for access points to Very High Radiation Areas; of structural shielding materials, including dimensions and specifications for materials used for shielding; to support inservice inspections of structures; provided to minimize contamination, to the extent practicable, of the facility or environment.

3.11 ENVIRONMENTAL QUALIFICATION OF MECHANICAL AND ELECTRICAL EQUIPMENT – as it relates to design features provided to control radiation exposure to components and SSCs in order to maintain the qualification of mechanical, electrical and electronic equipment.

4.2 FUEL SYSTEM DESIGN – as it relates to design features provided to minimize ORE from the fuel, such as cladding material and grid straps.

4.5.1 CONTROL ROD DRIVE STRUCTURAL MATERIALS – as it relates to the neutron absorber materials and fabrication design criteria, provided to reduce ORE.

4.5.2 REACTOR INTERNAL AND CORE SUPPORT STRUCTURE MATERIALS – as it relates to the specification of materials to reduce ORE, such as specifications for low cobalt content, the use of ORE reducing technologies, such as zinc injection, the types and methods of construction (e.g. the use of double wall pins versus single walled pins to improve integrity) of start-up neutron sources, the types of neutron detection equipment (e.g. in core neutron detectors) and features provided to minimize neutron irradiation of plant structures and components.

5.2.3 REACTOR COOLANT PRESSURE BOUNDARY MATERIALS – as it relates to the material specifications provided to reduce ORE (e.g. low cobalt content), chemistry controls to minimize corrosion and reduce ORE (e.g. Electric Power Research Institute (EPRI) primary water chemistry guidelines and zinc injection), thermal hydraulic design features provided to limit erosion (e.g. limiting flow rates, or the use of baffles) and fabrication techniques, such as processes to ensure smooth surfaces resistant to erosion or the deposition of material and to minimize contamination.

5.2.4 REACTOR COOLANT PRESSURE BOUNDARY INSERVICE INSPECTION AND TESTING – as it relates to the methods and features provided to reduce ORE due to inspections such as: reducing inspection frequencies, improving access to SSCs, using improved inspection techniques.

5.2.5 REACTOR COOLANT PRESSURE BOUNDARY LEAKAGE DETECTION – as it relates to the radiation monitoring systems, types of detectors and specified sensitivity, provided for leakage detection to ensure the integrity of the Reactor Coolant System (RCS), and to minimize contamination of the facility and reduce ORE through early detection of leaks.

5.3.1 REACTOR VESSEL MATERIALS - as it relates to the material specifications provided to reduce ORE (e.g. cobalt content).

5.4 REACTOR COOLANT SYSTEM COMPONENT AND SUBSYSTEM DESIGN – as it relates to the features provided to limit or reduce the build up of radioactivity in tanks, heat exchangers, and related components connected to the RCS), and features provided to limit ORE, minimize contamination of the facility and reduce waste generation from these potential radiation sources.

5.4.2.1 STEAM GENERATOR MATERIALS – as it relates to the materials (e.g. cobalt content) and design features (e.g. access for tube testing) provided to minimize ORE, minimize contamination of the facility and facilitate decommissioning.

5.4.7 RESIDUAL HEAT REMOVAL (RHR) SYSTEM – as it relates to the design features provided to minimize introduction of corrosion products into the RCS, features provided to minimize ORE from activity contents of the system, and minimize contamination of the facility and minimize leakage of radioactive fluids.

5.4.13 ISOLATION CONDENSER SYSTEM – As it relates to design features provided for leakage detection and prevention, features provided to minimize ORE from the system during operation and maintenance, and minimize contamination of the facility and minimize leakage of radioactive fluids.

BTP 5-1 MONITORING OF SECONDARY SIDE WATER CHEMISTRY IN PWR STEAM GENERATORS – as it relates to the sensitivity and detector types specified for radiation monitors provided for compliance with NEI 97-06 and the underlying EPRI Guidelines and to minimize contamination of the facility and reduce ORE through early detection of leaks.

6.1.1 ENGINEERED SAFETY FEATURES MATERIALS – as it relates to the design features provided to minimize introduction of corrosion products into the RCS, features provided to minimize ORE from activity contents of the system, and minimize contamination of the facility and minimize leakage of radioactive fluids.

6.2.1 CONTAINMENT SYSTEM FUNCTIONAL DESIGN – as it relates to the design features provided to minimize ORE due to operation, inspection and maintenance of containment SSCs.

6.2.2 CONTAINMENT HEAT REMOVAL SYSTEMS - as it relates to the design features provided: to minimize ORE due to operation, inspection and maintenance of containment heat removal SSCs; maintain ORE to personnel accessing containment ALARA; and to control radiation exposure in order to maintain the qualification of mechanical, electrical and electronic equipment.

6.4 CONTROL ROOM HABITABILTY SYSTEM – as it relates to the design features provided: to minimize ORE due to operation, testing and maintenance of control room ventilation system SSCs; shield operators from radiation exposure; and to protect control room operators from airborne radioactive materials.

6.5.3 FISSION PRODUCT CONTROL SYSTEMS AND STRUCTURES – as it relates to limiting or reducing radioactive fission, activation and corrosion product sources within the SSCs, features provided to reduce ORE during operation, testing, and maintenance of Engineered Safety Features (ESF) SSCs, and provisions to minimize contamination of the facility and minimize leakage of radioactive fluids.

6.6 INSERVICE INSPECTION OF CLASS 2 AND 3 COMPONENTS – as it relates to the methods and features provided to reduce ORE due to inspections such as: reducing inspection frequencies, improving access to SSCs, using improved inspection techniques.

7.1 INSTRUMENTATION AND CONTROLS – as it relates to the design features provided for radiological protection of plant workers, reducing ORE associated with servicing and maintaining of plant instrumentation used to minimize contamination of the facility and the environment.

7.3 ENGINEERED SAFETY FEATURES SYSTEMS – as it relates to the design features provided for radiological protection of plant workers during operation and reducing ORE associated with servicing and maintaining of equipment.

7.5 INFORMATION SYSTEMS IMPORTANT TO SAFETY – as it relates to the instrumentation provided for monitoring radiological conditions during an accident.

BTP 7-10 GUIDANCE ON APPLICATION OF REGULATORY GUIDE 1.97 – as it relates to types, ranges and qualification of radiation monitoring equipment required for accident monitoring.

9.1.1 CRITICALITY SAFETY OF FRESH AND SPENT FUEL STORAGE AND HANDLING – as it relates to the features provided to ensure adequate shielding and cooling of irradiated fuel and irradiated core components, within the refueling area, in transit and in spent fuel pool storage areas in order to maintain ORE ALARA.

9.1.3 SPENT FUEL POOL COOLING AND CLEANUP SYSTEM HANDLING – as it relates to the features provided to ensure adequate shielding and cooling of irradiated fuel and core components, within the refueling area, in transit and in spent fuel pool storage, and the shielding and design features of filtration and purification media of refueling and fuel storage pools provided to maintain ORE ALARA.

9.1.4 LIGHT LOAD HANDLING SYSTEM (RELATED TO REFUELING) - as it relates to the design features provided to ensure adequate shielding during storage, movement and handling of spent fuel and irradiated components in order to maintain ORE ALARA.

9.2.2 REACTOR AUXILIARY COOLING WATER SYSTEM – as it relates to the design features provided to minimize and reduce the amount of radioactive fission, activation and corrosion products contained within the system, reduce ORE during operation testing and maintenance, and minimize contamination of the facility and minimize leakage of radioactive fluids.

9.2.4 POTABLE AND SANITARY WATER SYSTEMS – as it relates to the design features provided to minimize and reduce the amount of radioactive fission, activation and corrosion products contained within the system in order to minimize contamination of the facility and minimize leakage of radioactive fluids.

9.2.6 CONDENSATE STORAGE FACILITIES – as it relates to the design features provided to minimize and reduce the amount of radioactive fission, activation and corrosion products contained within the system, reduce ORE during operation testing and maintenance, minimize contamination of the facility and minimize leakage of radioactive fluids and control exposure of members of the public to direct sources of radiation.

9.3.1 COMPRESSED AIR SYSTEM – as it relates to design features provided to prevent radiological contamination of the system. To the extent that the system is used as a source of breathing air, the design features provided to ensure protection of personnel including protection from contaminants, air quality monitoring and provisions for ensuring adequate air supply to respiratory protection equipment users.

9.3.2 PROCESS AND POST-ACCIDENT SAMPLING SYSTEMS – as it relates to design features provided to: minimize ORE during operation, AOOs and Design Basis Events (DBE); minimize waste generation; obtain representative samples; minimize contamination of the facility, and minimize leakage of highly radioactive fluids.

9.3.3 EQUIPMENT AND FLOOR DRAINAGE SYSTEM – as it relates to design features provided to minimize and remove sources of radiation (e.g. crud traps) and to minimize contamination of the facility and the environment.

9.3.4 CHEMICAL AND VOLUME CONTROL SYSTEM (PWR) (INCLUDING BORON RECOVERY SYSTEM) – as it relates to the minimization, reduction and shielding of radioactive fission, activation and corrosion products within the system piping tanks and vessels, including the associated filtration and purification media and features provided to minimize the introduction of material into the RCS, such as the specification of low cobalt content, equipment design features that limit erosion (e.g. smooth surfaces or design flow rates and baffles to help reduce erosion), features provided to minimize system leakage, and features provided to minimize required maintenance.

9.4.1 CONTROL ROOM AREA VENTILATION SYSTEM – as it relates to limiting or reducing radioactive fission, activation and corrosion product sources within the ventilation filtration media, tanks and structures, and providing features to reduce ORE during operation testing and maintenance of ventilation systems components, and providing features to minimize contamination of the facility.

9.4.2 SPENT FUEL POOL AREA VENTILATION SYSTEM – as it relates to limiting or reducing radioactive fission, activation and corrosion product sources within the ventilation filtration media, tanks and structures, and providing features to reduce ORE during operation testing and maintenance of ventilation systems components, and providing features to minimize contamination of the facility.

9.4.3 AUXILIARY AND RADWASTE AREA VENTILATION SYSTEM – as it relates to limiting or reducing radioactive fission, activation and corrosion product sources within the ventilation filtration media, tanks and structures, and providing features to reduce ORE during operation testing and maintenance of ventilation systems components, and providing features to minimize contamination of the facility.

9.5.2 COMMUNICATIONS SYSTEMS – as it relates to ensuring adequate communications are provided for the purpose of minimizing ORE within the radiologically controlled area during operation, AOOs and DBEs.

9.5.3 LIGHTING SYSTEMS – as it relates to design features for; providing adequate normal and emergency lighting in radiologically controlled areas during operation, AOOs and DBEs; providing features to minimize ORE associated with maintenance and servicing of normal and emergency lighting.

10.4.6 CONDENSATE CLEANUP SYSTEM - as it relates to limiting and reducing the radioactive fission, activation and corrosion products within the system piping tanks and vessels, including the associated filtration and purification media, providing design features to limit ORE during operation, testing and maintenance, and providing design features to minimize contamination of the facility.

10.4.8 STEAM GENERATOR BLOWDOWN SYSTEM - as it relates to limiting and reducing radioactive fission, activation and corrosion products within the system piping tanks and vessels, including the associated filtration and purification media, providing design features to limit ORE during operation, testing and maintenance, and providing design features to minimize contamination of the facility.

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11 RADIOACTIVE WASTE MANAGEMENT, as it relates to the description of the design features provided for containment, shielding and handling material contained within the radioactive waste management system, provisions for maintaining ORE ALARA during routine operation, AOO and DBEs, and providing design features to minimize contamination of the facility.

14 RADIATION PROTECTION – INSPECTIONS, TESTS, ANALYSES, AND ACCEPTANCE CRITERIA – as it relates to ITAAC for radiation protection equipment and equipment provided to reduce ORE.

16.0 TECHNICAL SPECIFICATIONS (TSs) – as it relates to identification of requirements for High Radiation Area and Very High Radiation Area access controls, and any requirements for radiation monitors described in Chapter 12.

II. <u>ACCEPTANCE CRITERIA</u>

<u>Requirements</u>

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

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

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

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

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

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

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

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

8. General Design Criterion (GDC) 19 found in Appendix A to 10 CFR Part 50, as it relates to the provision of adequate radiation protection to permit access to areas necessary for occupancy after an accident, without personnel receiving radiation exposures in excess of 50 millisievert (mSv) (5 rem) Total Effective Dose Equivalent (TEDE) as defined in 10 CFR 50.2, to the whole body or the equivalent to any part of the whole body for the duration of the accident in accordance with 10 CFR 50.34(f)(2)(vii)[1] and NUREG 0737, Item II.B.2.

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

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

11. 10 CFR 50.68~~, as~~ or 10 CFR 70.24as it relates to procedures and criteria for radiation monitoring in areas where special nuclear material is stored and handled.

12. 10 CFR 52.47(b)(1), which requires that a DC FSAR contain the~~ proposed~~ 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~~facility that incorporates the DC ~~is built~~has been constructed and will ~~operate~~be operated in accordance with the DC, the provisions of the Atomic Energy Act (AEA), and the U.S. Nuclear Regulatory Commission's (~~NRC's~~NRC) regulations;

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

14. 10 CFR 50.34(f)(2)(xvii) and NUREG 0737, Item II.F.1, which requires the applicant to provide instrumentation to monitor containment radiation intensity (high level).

15. 10 CFR 50.49(e)(4) and GDC 4 found in Appendix A to 10 CFR Part 50 which requires the determination of the radiation environment expected during normal operation and the most severe design bases accident, for electric equipment relied upon to remain functional during and following DBEs, including AOOs.

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

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16. 10 CFR 50.34(f)(2)(vii) and Section II.B.2 of NUREG-0737, 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.

17. GDC 14 and GDC 30 as they relate to the ability to detect RCS pressure boundary leakage with radiation detectors.

18. 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.

19. 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.

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

SRP Acceptance Criteria

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

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

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

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

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3. ~~3.~~ RG 1.7, as it relates to ~~methods~~protection from radionuclides in systems used for determining gaseous ~~radionuclides~~concentrations in containment following an accident.

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

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

6. ~~6. RG 1.97, Revision 4~~RG 1.97, BTP 7-10 and the Memorandum from D.G. Eisenhut, Nuclear Reactor Regulation (NRR), to Regional Administrators dated August 16, 1982, as it relates to a method acceptable to the staff for complying with the Commission’s regulations to provide and calibrate instrumentation for radiation monitoring following an accident in a light~~-~~-water~~-~~-cooled nuclear power plant.

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

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

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

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

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

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

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

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

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[3] ~~RG 1~~Regulatory Guide1.183 is applicable to applicants or license holders issued after January 10, 1997.

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15. ~~15.~~ NUREG-1433, as it relates to radiation protection considerations in the applicability, format, and implementation of the General Electric (BWR/4) Technical Specification package.

16. ~~16.~~ NUREG~~-~~-1434, as it relates to radiation protection considerations in the applicability, format, and implementation of the General Electric (BWR/6) Technical Specification package.

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

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

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

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

21. ~~21.~~ ANSI/ANS~~-~~-6.4-~~1997 (R2004),~~2006, as it relates to requirements and recommended practices for the construction of concrete radiation shielding structures.

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

23. RG 1.140, as it relates to actions taken to address the guidance contained in RG 8.8 Position C.2(d), during facility design, engineering, construction, and decommissioning to maintain ORE ALARA in accordance with 10 CFR 20.1101(b) and the definition of ALARA in 10 CFR 20.1003, concerning the radiation protection information to be supplied in SAR Section 12.

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

25. RG 4.21, as it relates to the design features provided to minimize contamination of the facility and environment, facilitate decommissioning and minimize the generation of radioactive waste.

26. RG 1.45, as it relates to the detection capabilities of radiation monitors described in Chapter 12 that are provided for RCS pressure boundary leakage detection, to the extent that they are not addressed in other section of the SRP.

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27. NEI 97-06, as it relates to the leakage detection capabilities of the radiation monitoring equipment described in Chapter 12 of the SAR, that are provided to detect steam generator tube leakage, in accordance with the criteria specified in the EPRI bases documents to the extent that they are not addressed in other sections of the SRP.

28. RG 1.143, regarding design features provided to minimize ORE and classification of structures housing radioactive waste systems based on potential exposure to site personnel.

29. BTP11-3 and SECY 94-198 as they relate to design features provided to minimize ORE for radioactive waste storage facilities described in the application.

The specific SRP acceptance criteria are:

1. Facility Design Features

The acceptability of the facility design features will be based on evidence that the applicant has fulfilled the dose limiting requirements of 10 CFR 50.34(b)(3), 52.47(a)(5), 10 CFR 52.79(a)(3) and 10 CFR 52.157(e), 10 CFR 20.1201, 10 CFR 20.1202, 10 CFR 20.1203, 10 CFR 20.1204, and 10 CFR 20.1207, as well as the radiation protection aspects of GDC 19 and 61, and 10 CFR 50.34~~.~~, and 10 CFR 50.49 for controlling radiation dose to electrical equipment important to safety and 10 CFR 52.47(a)(22). This includes evidence that major exposure accumulating functions (maintenance, refueling, radioactive material handling and processing, inservice~~-~~ inspection, calibration, decommissioning, and recovery from accidents) have been considered in plant design ~~and that~~that the evidence should also include radiation protection features incorporated into the design, taking into account the state of technology, that will keep potential radiation exposure from these activities ALARA in accordance with 10 CFR 20.1101(b), the definition of ALARA in 10 CFR 20.1003, and RGs 8.8 and 8.10. Such features may include (1) the ease of accessibility to work, inspection, and sampling areas, (2) the ability to reduce source intensity, (3) design measures to reduce the production, distribution, and retention of activated corrosion products~~,~~ (e.g. material selection, water chemistry and decontamination connections), (4) the ability to reduce time required in radiation fields, and (5) a provision for portable shielding and remote handling tools. Evidence of methods to control personnel exposure from high dose rate components such as temporary storage areas for irradiated fuel and irradiated core components (e.g. storage and handling of fixed in core detectors during outage) should be considered during plant design. Access control will be judged for acceptability in accordance with the requirements of 10 CFR 20.1601, 10 CFR 20.1602, 10 CFR 20.1901, 10 CFR 20.1902, and 10 CFR 20.1903 or access control alternatives in Standard Technical Specifications (NUREG-1430, NUREG-1431, NUREG-1432, NUREG~~-~~-1433, and NUREG-1434).

Facility design, to the extent practicable, should minimize the potential for creating a very high radiation area during normal operations, including ~~abnormal operational occurrences~~AOOs (such as dropping a fuel bundle during fuel handling operations). High and very high radiation areas should be remote from normally occupied rooms and corridors such that personnel access to these areas can be controlled in accordance

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with 10 CFR 20.1601 and 10 CFR 20.1602 and the guidance in RG 8.38. All accessible portions of the spent fuel transfer tube or canal that are capable of having radiation levels greater than 1 gray (Gy) per hour (100 rads per hour) should be shielded during fuel transfer. This shielding should be such that the resultant contact radiation levels are no greater than 1 Gy per hour (100 rads per hour). All accessible portions of the spent fuel transfer tube are clearly marked with a sign stating that potentially lethal radiation fields are possible during fuel transfer. If removable shielding is used for the fuel transfer tubes, it must also be explicitly marked as above. If other than permanent shielding is used, local audible and visible alarming radiation monitors must be installed to alert personnel if temporary fuel transfer tube shielding is removed during fuel transfer operations. Similar precautions should also apply to any other plant radiation source having radiation levels greater than 1 Gy per hour (100 rads per hour).

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

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

Consistent with the guidance contained in RGs 8.8 and 1.143, BTP 11-3 and SECY 94-198, “Review of Existing Guidance Concerning the Extended Storage of Low-Level Radioactive Waste,” SSCs that are described in the application, should be designed to control leakage and facilitate access, operation, inspection, testing, and maintenance in order to maintain radiation exposures to operating and maintenance personnel as low as is reasonably achievable. Structures housing radioactive waste processing systems or components should be classified using the guidance for potential exposure to site personnel, contained in RG 1.143.

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2. <u>Shielding</u>

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

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

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

3. <u>Ventilation</u>

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

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to personnel maintaining them, or those in adjacent occupied areas~~.~~, consistent with the guidance contained in the RG 8.8 and the requirements of 10 CFR 20.1101(b) and 10 CFR 52.47(a)(22). Acceptability of the ventilation system, relative to radioactive gases and particulates, will also be based on evidence that the applicant has applied the guidance of RG 8.8 and RG 1.140 or proposed acceptable alternatives.

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

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4. Area Radiation and Airborne Radioactivity Monitoring Systems

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A. The area radiation monitoring systems will be acceptable if they meet the provisions of 10 CFR 20.1501, 10 CFR 50.34(f)(2)(xvii); the guidance in NUREG-0737, RG 8.25,~~ and ~~RG 1.97, ~~Revision 4;~~BTP 7-10 and the following criteria:

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

i. ~~i.~~ Personnel have the capability to assess the radiation hazard in areas that may be accessed during the course of an accident, in accordance with the criteria of 10 CFR 50.34(f)(2)(xvii); NUREG-0737, ~~Item II.F.1; and RG 1.97, Revision 4~~item II.F.1; and RG 1.97 and the Memorandum from D.G. Eisenhut, NRR, to Regional Administrators dated August 16, 1982 regarding calibration of radiation monitoring equipment.

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

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

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

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

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

E. Instrumentation for monitoring areas where reactor fuel is stored or handled will be acceptable if it meets the criteria of 10 CFR 50.68 or 10 CFR 70.24.

F. To the extent that it is not covered in section application Section 5.2.5 or Section 11, the specified sensitivity of radiation monitoring equipment provided for Reactor Coolant Pressure Boundary Leakage detection (i.e. RG 1.45) is acceptable if it is capable of meeting the required fluid leakage detection criteria.

G. To the extent that it is not covered in BTP 5-1 or application Section 11.5, the specified sensitivity of radiation monitoring equipment provided for Primary to Secondary Leakage detection (i.e. requirements specified in the EPRI guidance forming the basis for NEI 97-06) is acceptable if it is capable of meeting the required fluid leakage detection criteria.

H. To the extent that it is not covered in application Sections 7.5 and 11.5, the description of the Emergency Response Data System is acceptable if the radiation monitoring system components required by 10 CFR Part 50. Appendix E VI.2(a) is provided.

5. Dose Assessment

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

If applicable, the applicant's dose assessment of construction workers on a facility adjacent to an existing nuclear unit(s), will be acceptable if it documents the assumptions made, calculations used, results for the areas where construction workers will be located (including numbers of construction workers), expected dose contributions (from direct, gaseous, and liquid sources), and projected person-Sievert (person-rem) doses, consistent with RG 1.206 Subsection C.I.12.3.5.

6. Minimization of Contamination

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Compliance with 10 CFR 20.1406 requires the applicant to describe how facility design and procedures for operation will minimize, to the extent practicable, contamination of the facility and the environment, facilitate eventual decommissioning, and minimize, to the extent practicable, the generation of radioactive waste. The acceptability of these features will be based on the guidance contained in RG 4.21 and Appendix 12.3-12.4-A (CP PSAR and updates in the OL FSAR, DC FSAR, or the COL FSAR to the extent that they are not addressed in a referenced certified design.)

Technical Rationale

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

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

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

3. The references to the specific items in 10 CFR 50.34(f), their associated action item in NUREG-0737, and RG 1.~~97, Revision 4,~~97and BTP 7-10 specify that adequate in-plant radiation monitoring is provided for accidents and ~~abnormal operational occurrences~~AOOs. Radiation protection design features are provided to allow personnel access to the plant under accident conditions sufficient to perform actions necessary to mitigate the consequences of the accident.

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

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

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

5. Compliance with the requirements of 10 CFR 50.68 or 10 CFR 70.24 and GDC 63 ensures that appropriate radiation monitoring is provided in areas of the plant where

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6. Compliance with 10 CFR 50.49(e)(4) and GDC 4 found in Appendix A to 10 CFR Part 50 ensures that the radiation environment expected during normal operation and the most severe design bases accident, will not exceed the functional capabilities of electric equipment relied upon to remain functional during and following DBEs, including AOOs.

7. Compliance with the requirement of Criterion 30-Quality of reactor coolant pressure boundary, ensures that means are provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

   Compliance with the requirements of 10 CFR Part 50, Appendix E VI.2(a), ensures the provision of accurate and timely radiation monitor data needed to determine core and coolant system conditions well enough to assess the extent or likelihood of core damage and to determine the conditions inside the containment building well enough to assess the likelihood and consequence of its failure.

8. Compliance with the requirements of 10 CFR 20.1406 in an early stage of planning ensures that the facility will be designed and operated, to the extent practicable, in a way that would minimize the contamination of the facility, contamination of the environment, and the generation of radioactive waste, and would facilitate decommissioning. 10 CFR 20.1406, applies to all DC and COL applications submitted after August 20, 1997.

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

III. REVIEW PROCEDURES

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

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

1. The staff will review the information on radiation protection design features furnished in the SAR, including referenced parts of Chapters ~~9 and 11~~3, 4, 5, 6, 7, 9, 10, 11, 13.4, 14.3.8 and 16, for completeness in accordance with RG 1.70 (or RG 1.206 for DC or COL applicants under 10 CFR Part 52). The reviewer will evaluate the SAR text and the scaled layout drawings of the facility, concentrating on the sources, shielding, and layouts for the

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Formatted: Indent: Left: 0", Hanging: 0.5", Widow/Orphan control, Tab stops: -1", Left auxiliary building, including the radwaste systems, decontamination facilities, office and access control areas, laundry, lockers and shower rooms (including the personnel decontamination area), and laboratory facilities; the fuel handling facilities, including the spent fuel pool fuel transfer and related equipment~~; and~~ the BWR turbine building, including location of steam lines, reheaters, ~~and~~ moisture separators, temporary fuel handling or storage locations, irradiated component handling and storage locations and other areas containing radioactive material or contributing to the radioactive content of plant systems. For the CP PSAR, this review is particularly concerned with preliminary design features that may not appear to be consistent with ensuring that ORE will be ALARA. This review will evaluate the radiation protection design features using the guidelines of RG 8.8. The EPRI developed the "Utility Requirements Document" (URD) for evolutionary and advanced light water reactor (LWR) designs based on proven technology of 40 years of commercial U.S. and international LWR experience. NUREG-1242 "NRC Review of Electric Power Research Institute's Advanced Light Water Reactor Utility Requirements Document," documented the NRC staff's safety evaluation of the URD. The URD reviewed by the staff in 1992 referenced a number of industry documents, such as NP- 6516, "Guide for the Application and Use of Valves in Power Plant Systems", NP-5479, "Application Guidelines for Check Valves in Nuclear Power Plants," NP-5697, "Valve Stem Packing Improvements," NP- 6737, "Cobalt Reduction Guidelines," and NP-6316, Guidelines for Threaded-Fastener Application in Nuclear Power Plants," that provided contemporary operating experience regarding design practices beneficial to reducing ORE. While the state of technology has advanced since the issuance of the initial URD, the reports referenced within the URD, revised versions of those reports and new reports (e.g. "Pressurized Water Reactor Primary Water Zinc Application Guidelines,") related to improving equipment reliability are sources of information that describe the current state of technology that may be used to evaluate design specifications provided to ensure ORE is ALARA through the use of reliable and low maintenance valves, pumps and other components, consistent with the guidance in RG 8.8 and the requirements of 10 CFR 20.1003 and 1101(b), and 10 CFR 52.47(a)(22) to ensure that operating insights have been incorporated into the plant design. The reviewer will consider plant layout and intended access and egress traffic patterns both to determine conformance with 10 CFR 20.1601, 10 CFR 20.1602, 10 CFR 20.1901, 10 CFR 20.1902, 10 CFR 20.1903, 10 CFR 20.1904, 10 CFR 20.1905, or Standard Technical Specifications and to determine whether they will control access properly in limited and restricted access areas (high radiation and very high radiation areas). The staff will review SAR Chapters 5, 9 and 11 as necessary to evaluate dose rates in and around the spent fuel pool areas, the location of airborne radioactivity monitoring instruments within ventilation systems, and radwaste systems as they relate to radiation protection design. The reviewer will evaluate all relevant aspects of the initial design plans, particularly to identify new arrangements, improved designs, unusual shield thicknesses, a new or modified shield thickness calculational procedure, unusual assumptions in the calculation, and placement of radiation monitors. The staff responsible for the review of SRP Chapter 11 will evaluate the adequacy of the process and effluent radiation monitoring (e.g., sensitivity, range, system placement) design.

2. RG 1.97~~, Revision 4,~~ and BTP 7-10 as referenced above, ~~provides~~provide detailed guidance and criteria for post accident radiation monitoring instrumentation. ~~Revision 4 to the RG provides a method for applying this guidance to the specific proposed design.~~ The staff will coordinate the review of the radiation monitoring systems with the

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instrumentation and control and emergency preparedness review staff to ensure that adequate radiation detection instrumentation is provided for plant monitoring under accident conditions.

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

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

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

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

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

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

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

9. The staff will review the information on design features furnished in the SAR provided to minimize contamination of the facility and environment, minimize the generation of radioactive waste and facilitate decommissioning, including referenced parts of Chapters 3, 4, 5, 6, 7, 9, 10, 11, 13.4, 14.3.8 and 16, for completeness in accordance with the guidance contained in RG 4.21 and Appendix 12.3-12.4-A (or RG 1.206 for DC or COL applicants under 10 CFR Part 52). The reviewer will evaluate the SAR text and the scaled layout drawings of the facility, for descriptions of the design features provided to minimize contamination and facilitate decommissioning.

10. Using the guidance contained in Appendix 12.3-12.4-A and RG 4.21, the staff will review information in the application provided to describe how facility design and procedures for operation will minimize, to the extent practicable, contamination of the facility and the environment, facilitate eventual decommissioning, and minimize, to the extent practicable, the generation of radioactive waste. The reviewer will use the information provided in Chapter 12 of the SAR, supplemented and complemented as necessary by information contained in those sections of the SAR that describe systems containing radioactive material during normal operations, AOOs and accident conditions.

IV. EVALUATION FINDINGS

The reviewer verifies that the applicant has provided sufficient information and that the review and calculations (if applicable) support conclusions of the following type to be included in the staff's safety evaluation report. The reviewer also states the bases for those conclusions.

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

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

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

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

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

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

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

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Formatted: Indent: Left: 0.5", Right: 0.63" name]. [List several examples of shielding design features used at plant.] Some of the criteria used by [utility] in locating penetrations in shield walls at [plant name] are [list several shield penetration location criteria used]. These shielding techniques are designed to maintain personnel radiation exposures ALARA, in accordance with the provisions of RGs 8.8 and 8.10, and are acceptable.

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

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

The applicant's area radiation monitoring system is designed to (1) monitor the radiation levels in areas where radiation levels could become significant and where personnel may be present, (2) alarm when the radiation levels exceed preset levels to warn of increased radiation levels, and (3) provide a continuous record of radiation levels at key locations throughout the plant. To meet these objectives, the applicant plans to use [number] area monitors located in areas where personnel may be present and where radiation levels could become significant. The area radiation monitoring system meets the criteria of 10 CFR 50.34(f)(2)(xvii), ~~Item~~10 CFR Part 50, Appendix E VI.2(a),, item II.F.1(3) of NUREG-0737, ~~and~~ RG 1.~~97, Revision 4,~~97and BTP 7-10 and is equipped with local and remote audio and visual alarms and a facility for central recording. [List examples of other area monitoring system features~~.]~~] The design objectives of the applicants' airborne radioactivity monitoring system are (1) to assist in maintaining occupational exposure to airborne contaminants ALARA, (2) to check on the integrity of systems containing radioactivity, and (3) to warn of unexpected release of airborne radioactivity to prevent inadvertent exposure of personnel. The applicant will install airborne radioactivity monitors in work areas where there is a potential for airborne radioactivity. These airborne radioactivity monitors have the capability to detect ~~derived air concentrations in air (DAC)~~DAC of the most restrictive particulate and iodine radionuclides in the area or cubicle of lowest ventilation flow rate within 10 hours(s) (usually

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denoted as 10 DAC~~-~~-hrs). The applicant will provide portable continuous air monitors when needed to monitor air in areas not provided with fixed airborne radioactivity monitors. All airborne and area radioactivity monitors will be calibrated periodically. [List examples of other airborne radioactivity monitoring features~~.~~]] The objectives and location criteria of [plant name] area and airborne radiation monitoring systems are in conformance with those portions of 10 CFR 20.1501; 10 CFR 50.34, 10 CFR 52.47, or 10 CFR 52.79; and 10 CFR 50.68 or 10 CFR 70.24, as well as RG 1.97, ~~Revision 4, and~~BTP 7-10and RG 8.8, related to radiation and airborne radioactivity monitoring.

The objective of the applicant's accident radiation monitoring system is to provide the capability to assess the radiation hazard in areas that may be occupied during the course of an accident. The installed instruments have emergency power supplies, and the portable instruments are placed to be readily accessible to personnel responding to an emergency. The systems are designed for use in the event of an accident in terms of usable instrument range, with appropriate margins for the accident source term and the environment the instrument can withstand, and meet the provisions of 10 CFR 50.34(f)(2)(xvii), ~~Item~~item II.F.1(3) of NUREG-0737, and RG 1.97~~, Revision 4.~~.

Instrumentation to monitor plant areas where fuel is handled and stored meets the criteria of 10 CFR 50.68 or 10 CFR 70.24 and GDC 63 in Appendix A to 10 CFR Part 50 and is acceptable.

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

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

For DC and COL reviews, the findings will also summarize the staff's evaluation of requirements and restrictions (e.g., interface requirements and site parameters) and COL action items relevant to this SRP section.

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

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V. IMPLEMENTATION

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

The provisions of this SRP section apply to reviews of applications submitted ~~6~~six months or more after the date of issuance of this SRP section, unless superseded by a later revision.

The referenced RGs and NUREGs contain the implementation schedules for conformance to parts of the method discussed herein.

VI. REFERENCES

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

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

3. ~~3.~~ 10 CFR Part 51, “Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions.”

4. 10 CFR Part 52, “Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants.”

~~4. RG 1.3, “Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors.”~~

~~5. RG 1.4, “Assumptions Used for Evaluation of the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors.”~~

~~6. RG 1.7, “Control of Combustible Gas Concentrations in Containment Following a Loss-of Coolant Accident.”~~

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

~~8. RG 1.69, “Concrete Radiation Shields for Nuclear Power Plants.”~~

~~9. RG 1.70, “Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants.”~~

~~10. RG 1.97, Revision 4, “Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident.”~~

5. ~~11.~~ 10 CFR PART 70, “Domestic Licensing of Special Nuclear Material.“

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

~~12. RG 8.8, “Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable.”~~

~~13. RG 8.10, “Operational Philosophy for Maintaining Occupational Radiation Exposures as Low as Is Reasonably Achievable.”~~

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

~~15. RG 8.25, “Air Sampling in the Workplace.”~~

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

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

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

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

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

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

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

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

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

7. ~~25. RG 1.206, “Combined License Applications~~ANSI/ANS-6.4-2006, “Nuclear Analysis and Design of Concrete Radiation Shielding for Nuclear Power Plants ~~(LWR Edition).”~~.”

8. ~~26.~~ BTP 11-3 “Design Guidance for Solid Radioactive Waste Management Systems Installed in Light-Water-Cooled Nuclear Power Reactor Plants”

9. EPRI “Pressurized Water Reactor Primary Water Zinc Application Guidelines.”

10. Memorandum from Larry W. Camper to David B. Matthews and Elmo E. Collins, “List of Decommissioning Lessons Learned in Support of the Development of a Standard Review Plan for New Reactor Licensing” (~~Agencywide Documents Access and Management System (~~ADAMS~~)~~ Accession No. ML062620355), October 10, 2006.

11. Memorandum from D.G. Eisenhut, NRR, to Regional Administrators, August 16, 1982, “Proposed Guidance for Calibration and Surveillance Requirements for Equipment Provided to Meet Item II.F.1, Attachments 1, 2, and 3, NUREG-0737,” with enclosures.

12. NUREG-0737, “Clarification of TMI Action Plan Requirements.”18. NUREG-1430, “Standard Technical Specifications for Babcock and Wilcox Plants.”

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

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

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

16. NUREG-1434, “Standard Technical Specifications for General Electric Plants, BWR/6.”19.ANSI/ANS/HPSSC-6.8.1-1981, “Location and Design Criteria for Area Radiation Monitoring Systems for Light Water Nuclear Reactors.”

17. NUREG/CR-3587, “Identification and Evaluation of Facility Techniques for Decommissioning of Light Water Reactors~~” (ADAMS Accession No. ML081360413).~~.”

18. NUREG-1394, Revision 1, “Emergency Response Data System (ERDS) Implementation.”

19. NUREG-1242 "NRC Review of Electric Power Research Institute's Advanced Light Water Reactor Utility Requirements Document, Evolutionary Designs (Vol. 2, Pt. 1, ADAMS Ascension Number: ML100430013 and, Vol. 2, Pt. 2, ADAMS Ascension Number: ML063620331) and Passive Plant Designs (Vol. 3, Pt. 1, ADAMS Ascension Number: ML070600372 and, Vol. 3, Pt. 2, ADAMS Ascension Number: ML070600373).

20. RG 1.21, “Measuring, Evaluating, and Reporting Radioactive Material in Liquid and Gaseous Effluents and Solid Waste.”

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

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

23. RG 1.7, “Control of Combustible Gas Concentrations in Containment Following aLoss-of-Coolant Accident.”

24. RG 1.52, “Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants.”

25. RG 1.69, “Concrete Radiation Shields and Generic Shield Testing for Nuclear Power Plants.”

26. RG 1.70, “Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants.”

27. RG 1.97 “Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants.”

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28. RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors."

29. RG 8.2, "Guide for Administrative Practices in Radiation Monitoring."

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

RG 8.10, "Operational Philosophy for Maintaining Occupational Radiation Exposures as Low as Is Reasonably Achievable."

~~**PAPERWORK REDUCTION ACT STATEMENT**~~

31. ~~The information collections contained~~

32. RG 8.19, "Occupational Dose Assessment in Light-Water Reactor Power Plants Design Stage Man-Rem Estimates."

33. RG 8.25, "Air Sampling in the Workplace."

34. RG 8.38, "Control of Access to High and Very High Radiation Areas of Nuclear Plants."

35. RG 1.140 "Design, Inspection, and Testing Criteria for Air Filtration And Adsorption Units of Normal Atmosphere Cleanup Systems In Light-Water-Cooled Nuclear Power Plants"

36. RG 4.21, "Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning."

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

38. RG 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)."

39. RG 1.89 Revision 1, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants"

40. SECY-04-0032, "Programmatic Information Needed for Approval of a Combined License Application Without Inspections, Tests, Analyses, and Acceptance Criteria"

41. SECY-94-198 "Review of Existing Guidance Concerning the Extended Storage of Low-Level Radioactive Waste," August 1, 1994

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Paperwork Reduction Act Statement

This Standard Review Plan ~~are covered by the~~ contains and references information collection requirements ~~of 10 CFR Part 50 and 10 CFR Part 52, and~~ that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.). These information collections were approved by the Office of Management and Budget, approval ~~number~~numbers 3150-0014, 3150-0011 and 3150-0151.

~~PUBLIC PROTECTION NOTIFICATION~~

Public Protection Notification

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~~**SRP Section**~~

Appendix 12.3~~–~~-12.4-A
~~"Radiation Protection Design Features"~~
Evaluation and Acceptance Criteria for 10 CFR 20.1406 to Support Design Certification and Combined License Applications

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Purpose:

The purpose of this guidance is to provide further clarification on the evaluation and acceptance criteria that will be used by NRC staff in reaching a reasonable assurance finding that a DC or COL applicant has complied with the requirements of 10 CFR 20.1406, which applies to all DC and COL applications submitted after August 20, 1997. The rule requires that applicants describe how they intend to minimize, to the extent practicable, the contamination of the facility, the contamination of the environment, and the generation of radioactive waste. Applicants are also required to describe how they will facilitate decommissioning of the facility. The intent of Section 20.1406 is to emphasize to a license applicant the importance, in an early stage of planning, for facilities to be designed and operated in a way that would minimize the amount of radioactive contamination generated at the site during its operating lifetime and would minimize the generation of radioactive waste during decommissioning of the facility. Specific minimization requirements are directed towards those making an application for a new license because it is more likely that consideration of design and operational aspects that would reduce dose and minimize waste can be cost-effective at that time.

RG 4.21 describes a basis acceptable to the staff for implementing the requirements of 10 CFR 20.1406. This includes a discussion of high level objectives as well as specific actions that can be taken during design, construction, operation, and decommissioning to ensure that, to the extent practicable, contamination of the facility and the environment is minimized, radioactive waste generation is minimized, and decommissioning is facilitated.

Evaluation Criteria

If a COL application references a standard DC that meets the requirements of 10 CFR 20.1406 for design features, then the COL applicant needs to only consider those RG 4.21 criteria affecting operation and site-specific design features. At a minimum, as part of the description of design and operational features for all applicable SSCs, the applicant should also describe plans for: limiting leakage, controlling the spread of contamination, detecting leaks early, allowing for appropriate and timely action to mitigate and control the spread of contamination by the future licensee, and reducing the time, effort and hazard to personnel during decommissioning activities. Where appropriate to the type of SSC being considered, the applicant should explicitly describe how these considerations are addressed in the design and operation of the SSC.

General guidance on meeting the requirements of 10 CFR 20.1406 and examples have been developed and are included as Attachment A – "Evaluation and Scoping Information for Systems, Structures and Components 10 CFR 20.1406 Design Review" to this appendix. Attachment A provides scoping information for SSCs to assist the staff in evaluation of SSCs having a potential to release radioactive materials to the facility, site, or environment which could contaminate the soil or groundwater. In addition, Attachment C cites operational experiences for various SSCs, including actual Event Notices and information included in the Liquid Radioactive Release Lessons Learned Task Force (ADAMS Accession No. ML062650312).

Regulatory positions C.1 through C.4 in RG 4.21 are provided as specific guidance to applicants on meeting the requirements of 10 CFR 20.1406. C1 through C4 describe concepts to be implemented to provide reasonable assurance that inadvertent spills, leaks, and discharges of liquid, gaseous and solid radioactive effluents are prevented, detected and corrected, that the site is adequately characterized and understood, that decommissioning is planned for, and that the generation of radioactive waste is minimized. The measures to be taken by the applicant should be risk-informed and the examples described in Appendix A of RG 4.21, should be used by the applicants as guidance to determine which measures are applicable to their facility. Appendix A of RG 4.21 however, is not intended to be used as a checklist of minimally acceptable design or operational features. Alternative methods to RG 4.21 may be acceptable to meet the requirements of 10 CFR 20.1406, provided the methods are documented fully in the DC or the COL applications, and accepted by the staff.

Additionally, the applicant should document that if a spill, leak, or inadvertent discharge were to occur, design or operational features would ensure that the spill, leak, or discharge will be detected promptly, and monitored and evaluated to determine the impact on the environment.

Acceptance Criteria

To determine an applicant's compliance with 10 CFR 20.1406, as it relates to describing a basis acceptable for implementing the requirements of 10 CFR 20.1406, the staff should review the applicant's description of all applicable SSCs and applicable site-specific data against the guidance contained in RG 4.21 to confirm that:

- Adequate design features exist, supplemented with operating programs, processes and procedures (as necessary), and these will provide reasonable assurance that spills, leaks, and inadvertent discharges of radioactive effluents will be prevented to the extent practicable, or minimized.

- In the event the spill, leak, or inadvertent discharge does occur, the staff should verify that there is reasonable assurance that it will be detected in a timely manner. For those SSCs that are typically inaccessible for routine inspection or observation, leak detection capability, to the extent practical, should allow for the identification and measurement of relatively small leak rates, depending on the concentration (e.g. several gallons per week).

- Design features should be supplemented, as necessary, by operating programs, processes and procedures to monitor spills and leaks and evaluate their impact to the environment.

- The site has been adequately characterized and conceptual site models have been developed which define the site hydro geological setting including subsurface and surface migration pathways under both pre-construction and post-construction conditions. These models are needed to assist with designing monitoring components and procedures, designing protective measures, carrying out remediation, and designing decommissioning activities.

- Design features that facilitate decommissioning (such as modular components and adequate space for equipment removal) should be described, and their role in the decommissioning process should be described. Operating procedures to minimize the amount of residual radioactivity that will require remediation at the time of decommissioning should also be described.

- The site has been designed and will be operated to minimize the generation and volume of radioactive waste, both during operation and during decommissioning.

The NRC staff's safety evaluation report (SER) related to NEI technical report NEI 08-08A "Guidance for Life Cycle Minimization of Contamination" (ADAMS Accession Number ML093220530) provides the bases for the use of the template to describe an acceptable operational Ground Water Protection program which conforms to the guidance of RG 4.21. For those licensees that elect to demonstrate compliance with the programmatic requirements of 10 CFR 20.1406 via alternate methods, SECY-04-0032, "Programmatic Information Needed for Approval of a Combined License Application Without Inspections, Tests, Analyses, and Acceptance Criteria" notes that in the absence of ITAAC, "fully described" should be understood to mean that the program is clearly and sufficiently described in terms of the scope and level of detail to allow a reasonable assurance finding of acceptability at the COL stage.

References:

1. 10 CFR 20.1406, "Minimization of Contamination."

2. RG 4.21, "Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning." (ADAMS Accession No. ML080500187)

Appendix 12.3-12.4-A Attachment A

Evaluation and Scoping information for Systems, Structures and Components 10 CFR 20.1406 Design Review

I. General Guidance

Perform an evaluation of SSCs that contain or could contain radioactive liquids or material. Those SSCs that have a potential to release radioactive materials to the facility, site, or environment which could contaminate the soil or groundwater should be evaluated.

The regulations require that both design and operational processes be addressed. RG 4.21, "Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning," describes an acceptable method for applicants to meet the regulation. RG 4.21 also includes a list of examples that may be used to determine areas to address.

Examples of SSCs, such as those listed in Attachment B of this appendix, include, but are not limited to, radioactive waste systems, building sumps and drains, spent fuel storage pools and other systems where, based on operational experience, the likelihood of such releases could occur. Typical systems and operational experience instances are included in Attachment C of this appendix.

II. General SSC Screening

If the general screening indicates review is warranted, review the FSAR description provided to determine if the applicant has included design or operational features to address compliance with 10 CFR 20.1406. Request additional information or discuss with the Radiation Protection Branch responsible for reviewing SRP section 12.3, if additional information is needed.

1. Systems/Components:

a. Does the system contain or potentially contain *radioactive materials*? (See RG 4.21, Appendix A for examples)

AND;

b. Is the system separated from the environment by a single barrier?

- Tank/Sump with an exterior wall or floor
- Single walled pipe located in an area not accessible for inspection (buried pipe trench, pipe drains, etc.)

OR;

c. Are portions of the system located outside of a structure designed to contain a release of *radioactive materials*?

OR;

d. Has operational experience demonstrated that the system or components has previously resulted in a release of *radioactive materials?*

2. Structures:

a. Does the structure envelope a system or components that contain or potentially contain *radioactive materials*?

AND;

b. Are there any below grade penetrations (e.g., piping, conduit) to the environment?

OR;

c. Are there any below grade concrete joints (e.g., floor to floor, walls to floor) that connect to the environment?

OR;

d. Does the structure contain *radioactive materials* that are separated from the environment by a single barrier? (retention pond with liner, or radioactive waste pipe running between buildings)

Appendix 12.3-12.4-A Attachment B

Examples of Systems, Structures, and Components for 20.1406 Review

The list below provides examples of typical SSCs that typically have a potential to release radioactive material to the facility, site, or environment. Additional operating experience is provided as background information. This list is not intended to be complete and comprehensive, nor is it intended to be a checklist of minimally acceptable facility design features.

- Spent Fuel Storage and Transfer Systems
  - Spent Fuel Pool (SFP)
  - SFP Transfer Canal
  - SFP Leak Detection System

- Tanks and Piping
  - Radioactive Waste Tanks and Piping
  - Condensate Storage Tank and Piping
  - High-Pressure Coolant Injection (HPCI) and Emergency Service Water Piping
  - Refueling Water Storage Tank
  - Service Water and Component Cooling
  - Auxiliary Steam Lines
  - Cooling Tower Blowdown Line
  - Circulating water system piping
  - Retention Tanks
  - Discharge Canals and Piping (including air relief valves on lines)

- Drains
  - Water Treatment System Drains
  - Floor and Roof Drains
  - Laundry System Drains
  - Contaminated Sink Drains

  - Secondary Systems
  - Plant Chilled Water System
  - Cooling Tower Basin

  - Radioactive Waste Systems
  - Waste Disposal System Valves
  - Resin Fill Valve
  - Retention Ponds

  - Building
  - Building Sumps
  - Seismic Gaps
  - Joints

Appendix 12.3-12.4-A Attachment C
Operating Experiences for Review

SSC	Occurrence	Problem
Piping		
Non-safety, HPCI suction and return piping	Underground Pipe Leakage	Inadequate pipe design/maintenance
Condensate Tank, Condensate transfer system (underground pipe)	Degraded pipe caused leak. Liquid traveled outside the protected area via an underground telephone cable conduit run.	Inadequate pipe design/maintenance
Turbine Building Sump Discharge Line	Frozen end of discharge line caused liquid to backup and leak.	No freeze protection
Radioactive Waste liquid Effluent release pipe	Degraded effluent line piping	Inadequate pipe design
Coolant Tower Blowdown Line	Cooling tower blowdown line leak due to failure in piping.	Inadequate pipe design
Turbine and Waste Treatment Building Sump Discharge Line	Line leaked due to degraded condition of pipe	Inadequate pipe design
Underground pipe containing Uranium Bearing Discharge	Pipe ruptured underground and might have been undetected for years.	Inadequate pipe design/maintenance
Sumps		
Clean Sumps	Steam Leaks condensed and ran into clean sumps which were routed to storm drain pond.	Inadequate maintenance
Steam Lines		
Auxiliary Steam Lines	Steam and liquid leaks through seals, joints, and degraded pipes.	Inadequate maintenance
Retention Ponds		
Unlined Storm Drain Stabilization Pond (SDSP)	Tritium was found in two man holes located close to an unlined SDSP. The storm drain collector basin received overflow from the Turbine Building air-wash system, which contained small amounts of tritium	Inadequate design

SSC	Occurrence	Problem
Operating Practices		
Boric acid concentrator system (evaporator system) releases	Past operational practices during releases during rainy days from the system resulting in rain deposition and wash down of roof drains.	Inadequate operator procedures
Condensate transfer System	Liquid discharged from circulating water discharge tunnel via fire protection system and a portion of the service water system due to operator error.	Procedure Compliance
Outdoor storage of contaminated equipment	Contamination leached from equipment onto soil.	Inadequate procedures or operational controls
Condensate Storage Tank	Water overflowed from Condensate Storage Tank into a tunnel. Tunnel had potential to allow small amount of this water to permeate into the ground.	Inadequate procedures
Retention tank containing radioactive liquid	Retention tank containing uranium bearing liquid overflowed onto soil. Tank was undergoing maintenance and was not tight at the time.	Inadequate maintenance
Equipment		
Circulating Water Blowdown Line	Vacuum Breakers in blowdown line leaked while radioactive liquid traveled down the pipe.	Inadequate maintenance
Flange in feed water system venturi	Leak in system. Under drain system captured most of tritium from leakage.	Inadequate design/maintenance.
Steam Generator Tube leak	Liquid leaked from degraded Steam Generator Tube.	Inadequate design/maintenance
Tanks		
Storm Drains around Liquid Waste Holdup Tank	Liquid leaked through cracks in the asphalt berm around a Liquid Waste Holdup Tank Area into the groundwater.	Inadequate design/maintenance

SSC	Occurrence	Problem
Fuel Storage and Handling		
SFP	Estimated 141,500 gallons of SFP water was released in the gap between two reactor buildings into other buildings and surrounding environment. Operational/configuration control errors resulted in deflation of SFP seals and resultant leak.	Inadequate design and operational/configuration control error.
SFP	Liner leakage and hairline crack in Fuel Storage Building wall.	Inadequate design, bad weld
SFP	Failure of curtain drain	Inadequate design
SFP	SFP water leaked into narrow seismic gap due to clog in tell-tale drain system.	Inadequate maintenance
SFP	Defect in liner of cask loading pool resulted in leakage from cask loading pool.	Inadequate design
SFP Transfer Sleeve	Leakage through fuel transfer sleeve into abandoned Unit 2 facilities.	Inadequate design

Paperwork Reduction Act Statement

This Standard Review Plan contains and references information collection requirements that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.). These information collections were approved by the Office of Management and Budget, approval numbers 3150-0014, 3150-0011 and 3150-0151.

Public Protection Notification

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SRP SECTION 12.3-12.4
"RADIATION PROTECTION DESIGN FEATURES"
Description of Changes

~~This revision was necessitated to update to reference Revision 4 of RG 1.97. This action is acknowledged in Revision 4 of RG 1.97 and recommended only to provide consistency in SRP guidance. See ADAMS Accession No. ML112490115, dated September 23, 2011.~~

~~Revision 4 to SRP Section 12.3-12.4 updates Revision 3 of this section, dated March 2007 (ADAMS Accession No. ML070720019), to reflect the updated reference, i.e., Revision 4 of RG 1.97 throughout the SRP section and to Section VI. REFERENCES, Item 10.~~
This SRP section affirms the technical accuracy and adequacy of the guidance previously provided in Revision 4, dated March 2011 of this SRP. See ADAMS Accession No. ML113081427.

The technical changes incorporated in Revision 5 include: (1) incorporation of regulatory requirements of 10 CFR 20.1406 and the associated guidance contained in RG 4.21 (ML080500187) concerning the minimization of contamination and radioactive waste generation, (2) incorporation of Interim Staff Guidance DC/COL-ISG-6 "Evaluation and Acceptance Criteria for 10 CFR 20.1406 to Support Design Certification and Combined License Applications" (ML092470100) (3) incorporation of the requirements of 10 CFR 50.49(e)(4) and the guidance of RG 1.89 and RG 1.183 to control the radiation environment of equipment, (4) updated references to regulatory guidance documents, (5) incorporation of guidance to consider the bases for specifying sensitivity of radiation monitors described in this section of the SAR, that are provided for reactor coolant pressure boundary leakage detection, consistent with GDC 14 and GDC 30, (6) incorporation of guidance regarding the review of radiation monitoring equipment specified in 10 CFR Part 50, Appendix E, Subsection VI.2(a)(i), (6) incorporation of the regulatory bases stated in 10 CFR 52.47(a)(22), for including operating experience in the design phase, consistent with the existing requirements of 10 CFR 20.1101(b) and the definition of ALARA, and the guidance contained in RG 8.8, (6) incorporated use of the types of documents discussed in the EPRI URD and the associated Safety Evaluation documented in NUREG-1242, as sources of information available to the staff for assessing the use of the current state of technology to reduce ORE, consistent with the guidance contained in RG 8.8 and the requirements of 10 CFR 20.1101(b). The changes to this SRP Subsection reflect the experience gained by the staff developed during NRC reviews of DC and COL applications completed after Revision 4 of SRP 12.3-12.4 was issued.

The technical changes in each SRP section are as follows:

I. AREAS OF REVIEW

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

2. Added a statement regarding the use of RG 4.21 and Appendix 12.3-12.4-A guidance during 10 CFR 20.1406 design reviews.

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3. Added clarification regarding the use of guidance contained in RG 1.206, Subsection C.I.12.3.5, to provide a description of the basis for the dose assessment process for plant construction workers.

4. In order to improve how the staff finds information related to the design features discussed in RG 8.8 Regulatory Position C.2, RG 1.206 Subsection C.I.12.3 and RG 4.21, the discussion under Review Interfaces was expanded to provide

guidance to the staff on the sections of the SAR expected to identify the types and quantities of radioactive material in the plant.

II. ACCEPTANCE CRITERIA

1. Added clarification of the requirement to identify radiation monitors used to monitor high radiation levels in containment, consistent with 10 CFR 50.34(f)(2)(xvii) and NUREG-0737, Item II.F.1

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

3. Provided clarification regarding the need to ensure adequate access to important areas following an accident, consistent with 10 CFR 50.34(f)(2)(vii) and Section II.B.2 of NUREG- 0737.

4. Added a statement clarifying the need to ensure radiation monitors were provided to assess RCS pressure boundary leakage, consistent with GDC 14 and GDC 30.

5. Added a statement clarifying the requirement 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.

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

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

8. Added a statement clarifying the intended use of RG 1.70 during this review.

9. Added a statement clarifying the use of the guidance in RG 1.140 in conjunction with RG 8.8 guidance for evaluating design features to maintain ORE ALARA.

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

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11. Added statements regarding the detection capabilities of radiation monitoring equipment provided for RCS pressure boundary leakage detection in accordance with GDC 30 and the guidance contained in RG 1.45.

12. Added a statement regarding the detection capabilities of radiation monitoring equipment provided for primary-to-secondary leakage detection in accordance with the guidance contained in NEI 97-06.

13. Added a statement regarding the use of the guidance contained in RG 1.143 to assess the adequacy of design features to minimize ORE to site personnel and the classification of structures housing radioactive waste systems based on potential exposure to site personnel.

14. Added a statement regarding the use of the guidance contained in BTP11-3 and SECY 94-198 as they relate to design features provided to minimize ORE for radioactive waste storage facilities described in the application.

15. Added a statement to ensure the review included features provided to control ORE from irradiated material located in temporary outage storage locations, consistent with 10 CFR Part 20, Subpart G, RG 8.8 and RG 8.38.

16. Added clarification regarding the evaluation of radiation zones consistency with assumed source terms and shielding.

17. Added clarification regarding the evaluation of radiation monitoring equipment provided to meet 10 CFR 50.34(f)(2)(xvii), NUREG-0737, item II.F.1 and RG 1.97.

18. Added a statement regarding the description of radiation monitoring system components required by 10 CFR Part 50, Appendix E VI.2(a).

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

III. REVIEW PROCEDURES

1. Added a discussion regarding the review of design features for outage storage locations for irradiated components.

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

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3. Added clarification to include the use of BTP 7-10 in addition to RG 1.97 when reviewing some radiation monitoring equipment.

4. Added clarification to include review of potential high dose rates areas that may require personnel access during operation.

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

6. Added clarification regarding the review of the design features provided to comply with 10 CFR 20.1406.

IV. EVALUATION FINDINGS

1. Added references to 10 CFR Part 50, Appendix E VI.2(a), 10 CFR 70.24 and BTP 7-10.

V. IMPLEMENTATION

No Changes

VI. REFERENCES

1. Updated the descriptions of the existing references.

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

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