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Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Attention: Mr. Jeffrey A. Ciocco

Docket No. 52-021
MHI Ref: UAP-HF-10225

Subject: MHI's Response to US-APWR DCD RAI No. 578-4483 Rev. 2

References: 1) "Request for Additional Information No. 578-4483 Revision 2, SRP Section: 12.03-12.04 – Radiation Protection Design Features, Application Section: 12.3," dated April 27, 2010

With this letter, Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") a document entitled "Response to Request for Additional Information No.578-4483 Revision 2".

Enclosed are the responses to three RAIs contained within Reference 1.

Please contact Dr. C. Keith Paulson, Senior Technical Manager, Mitsubishi Nuclear Energy Systems, Inc. if the NRC has questions concerning any aspect of the submittal. His contact information is below.

Sincerely,

Y. Ogata

Yoshiki Ogata
General Manager- APWR Promoting Department
Mitsubishi Heavy Industries, LTD.

Enclosure:

1. Response to Request for Additional Information No.578-4483 Revision 2

CC: J. A. Ciocco
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Enclosure 1

UAP-HF-10225
Docket No. 52-021

Response to Request for Additional Information
No. 578-4483 Revision 2

July, 2010

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

7/30/2010

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 578-4483 REVISION 2**
SRP SECTION:	**12.03-12.04 – Radiation Protection Design Features**
APPLICATION SECTION:	**12.3**
DATE OF RAI ISSUE:	**4/27/2010**

QUESTION NO. : 12.03-12.04-37

20.1406, ISG-006, RG-4.21

US-APWR DCD RAI 91-1496, Question 12.03-12.04-2 asked the applicant to address compliance with 10 CFR 20.1406(b), which requires the application to describe how facility design and operation will minimize contamination of the facility and environment.

The applicant noted in their response that Design Control Document (DCD) Revision 0 was submitted to NRC, prior to the issuance of RG 4.21, so the design features to address the RG 4.21 specific objectives of minimization of contamination and radioactive waste generation during the design life of the plant and its subsequent decommissioning, were either expressly described in the various DCD sections on plant systems, or deferred for further design development with vendor specifics.

As noted in Interim Staff Guidance DC/COL-ISG-06 "Evaluation and Acceptance Criteria for 10 CFR 20.1406 to Support Design Certification and Combined License Applications" (ISG-6), 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. In Question 12.03-12.04-2 the applicant was asked to describe the DCD design features provided to minimize contamination and to provide COL Information items in the applicable sections of the DCD addressing the objectives which are more operational or procedural in nature for the specific systems. However, since FSAR Tier 2 Revision 2 Chapters 3, 5, 6 and 10 do not mention 20.1406, and the only reference to 10 CFR 20.1406 in Chapter 9 is to the Spent Fuel Pool leakage detection system, it is not clear to the staff that the applicant has fully described the design provisions and program element requirements provided for compliance with 10 CFR 20.1406.

In Question 12.03-12.04-2, the applicant was asked to provide the general description of design features and program elements provided to comply with 20.1406(b), in Section 12.3 of the DCD along with a pointer directing the reader to the section of the DCD which specifically addresses the system. This information would facilitate NRC Staff understanding of how the applicant complied with 20.1406. However, while the response to question 12.03-12.04-2 provided a summary of how some specific objectives are met on a system basis, it did not include this information in FSAR.

Supplemental Question:
Please revised and update the US-APWR FSAR Tier 2 Revision 2, to fully describe how facility design will minimize, to the extent practicable, contamination of the facility and the environment compliant with the requirements of 10 CFR 20.1406.

ANSWER:

The US-APWR design includes design features in conjunction with operational programs to address the requirements of 10 CFR 20.1406 and RG 4.21 by applying a contaminant management philosophy to the design of structures, systems, and components (SSC) that have the potential to contain radioactive materials. The application of the contaminant management philosophy produces a design that maintains occupational doses ALARA, minimizes environmental contamination, and facilitates the eventual decommissioning of the facility. The principal design features include one or more of the approaches listed below.

- Minimizes leaks through proper selection of materials compatible with the fluids and construction techniques (e.g., butt weld, floor sloping, etc)
- Detects leaks early through sloping of coated floor and collection of drains in leak detection system. Drainage piping should be kept in open coated concrete trenches as much as practicable in order to minimize embedded pipes.
- For tanks that contain radioactive fluids and are located outside, containment walls and berms with coating or liners, and sumps and level switches are provided to detect leaks and overflows. The design and construction of the tank foundations and berms are designed to comply with federal, state and local regulations. Tanks also have level instruments that can be used to indicate unintended releases.
- For waste minimization, the design provides multiple tanks for waste segregation and for separate processing; maximizes use of polished steel surface to facilitate ease of decontamination, and maximizes recycle and re-use of quality water. As part of the waste minimization effort, operation shall use only approved and non-hazardous chemicals to minimize the generation of mixed waste;
- To avoid unintended and unmonitored releases, building floors are slightly sloped to facilitate collection of drainage for early detection and treatment. In areas where the potential exists for spreading contamination, berms are provided to prevent spread of contamination and releases. For seismic category I and II structures, thick reinforced concrete with special porosity-reducing additives, waterproof coatings, and special sealing of joints and penetrations is used to reduce any unintended releases.
- For piping that carries contaminated or potentially contaminated fluid outside the buildings, the design emphasizes containment of leaks, prompt detection of leaks, provisions for inspections and mitigation for unintended releases. The piping design consists of single-walled piping enclosed within pipe chases or coated trench, or buried double-walled piping. Inspection manholes and leak detection instruments are provided for double-walled piping. The main trench is designed to be equipped with removable but sealed covers to minimize infiltration of precipitation, and to provide the capability to maintain, repair, and replace piping and trench when required. The trench also has inspection manholes with drain collection basins and liquid level switches. When fluid is detected, the instrument initiates alarms for operator actions that include inspection, sampling and analysis to determine the characteristics and source of the fluid, and extraction of the fluid for treatment and disposal. The design is supplemented by operational programs which includes periodic (once every two years) hydrostatic or pressure testing of pipe segments, instrument calibration, and when required, visual inspection and maintenance of piping, trench and instrument integrity. The inspection manholes, leak detection instruments, and testing of the piping minimize the need to remove the trench covers which helps maintain the integrity of the seals. Sections of the covers are removed only when leakage is detected or when trench repair is needed. For piping between buildings, penetration sleeves are provided to collect and direct any leakages back into the building floor drain system for further processing. For those situations in which the above containment approaches are not practicable and/or desirable, COL Item 12.3(10) is included to provide flexibility for the COL Applicant to implement designs and/or operational programs to comply with RG 4.21.

A new section 12.3.1.3 "Minimization of Contamination and Radioactive Waste Generation" has been added to the DCD to delineate the design objectives incorporated into the US-APWR, and includes a new Table 12.3-8 "Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste". The new table identifies the system design features to address the specific design requirements, stipulated in RG 4.21, and where these features are specifically discussed in the US-APWR DCD. This table is similar to the table provided in response to RAI 91-1496, Revision 1 dated January 9, 2009 (UAP-HF-09003, ML090150211), but expanded to include additional systems and design features. Also, a new Subsection 12.3.1.3.2 "Operational / Programmatic Considerations" has been added to the DCD. Additionally, COL action item 12.3(10) has been added to DCD Subsection 12.3.6, requiring the COL Applicant to address the site-specific, operational and post-construction objectives and conceptual site model guidance of Regulatory Guide 4.21.

NRC summarized the RG 4.21 requirements into 9 design and operational objectives in RAI 91-1496, Question 12.03-12.04-2. These 9 Objectives are captured below for convenience. Guided by these 9 objectives, a review of the various DCD chapters that are impacted by this and other related RAIs was conducted and cross-references to new Subsection 12.3.1.3 are added to the affected DCD subsections and system descriptions. Specifically, cross-references are added to Chapters 1, 3, 4, 5, 9, 10, and 11, covering the systems and subsystems with design features implemented to address 10 CFR 20.1406 and RG 4.21.

DCD Chapter 6 includes descriptions of Engineered Safety Features in terms of their ability to limit fission product releases to the environment, for example

- 6.2.1.6 – Leak testing of refueling water storage pit (RWSP) liner and containment liner
- 6.2.4 – containment isolation provisions to prevent or limit the release of fission products to the environment
- Table 6.3-1, item III.D.1.1 – leak detection and control for radioactive fluids outside containment

Although these capabilities are consistent with 10 CFR 20.1406 and RG 4.21, these ESF design features are governed by regulations and guidance specific to ESF functional performance requirements, e.g., General Design Criteria 16 and 50, and Item II.D.1.1 of NUREG-0737, in lieu of 10 CFR 20.1406 and RG 4.21. Hence, specific cross-references to RG 4.21 objectives are not necessary to demonstrate the ESF performance requirements. However, it is noted that some of the specific ESF that interface with the environment are discussed in more details in other sections of the DCD, such as 9.4.5 on HVAC systems, and cross-references to RG 4.21 are provided in these sections.

In conjunction with this new subsection, COL action items 12.1(6), 12.1(7), 12.1(8), and 12.3(10) address the following operational programs:

- COL Action Item 12.1(6) is to conduct periodic review of overall operational practices to ensure configuration management, personnel training and qualification update, and procedure adherence. This item addresses RG 4.21 Objective 5.
- COL Action Item 12.1(7) is to track implementation of requirements for record retention. This item addresses RG 4.21 Objective 6a and 6c.
- COL Action Item 12.1(8) is to supplement Subsection 12.3.1.3 for the development of procedures to limit leakage and the spread of contamination within the plant. This item, in conjunction with Subsection 12.3.1.3, addresses RG 4.21 Objectives 1, 2, 3, 4, 6b, and 7.
- COL Action Item 12.3(10) is to supplement Subsection 12.3.1.3 to address the site-specific design features, operational, post-construction objectives and conceptual site model guidance of RG 4.21. This development of a site model is to aid in the understanding the interface with environmental systems and the features that will control the movement of contamination in the environment. This item addresses RG 4.21 Objectives 8 and 9.

Summary of RG 4.21 design and operational objectives as stated in RAI 91-1496:

1) *Minimize leaks and spills and provide containment in areas where such events may occur,*
2) *Provide for adequate leak detection capability to provide prompt detection of leakage for any structure, system, or component which has the potential for leakage,*

3) *Use leak detection methods (e.g., instrumentation, automated samplers) capable of early detection of leaks in areas where it is difficult or impossible to conduct regular inspections (such as for spent fuel pools, tanks that are in contact with the ground, and buried, embedded, or subterranean piping) to avoid release of contamination from undetected leaks and to minimize contamination of the environment,*
4) *Reduce the need to decontaminate equipment and structures by decreasing the probability of any release, reducing any amounts released, and decreasing the spread of the contaminant from the source,*
5) *Periodically review operational practices to ensure that, operating procedures are revised to reflect the installation of new or modified equipment, personnel qualification and training are kept current, and facility personnel are following the operating procedures,*
6) *Facilitate decommissioning by a) maintenance of records relating to facility design and construction, facility design changes, site conditions before and after construction, onsite waste disposal and contamination and results of radiological surveys, b) minimizing embedded and buried piping, and c) designing the facility to facilitate the removal of any equipment and/or components that may require removal and/or replacement during facility operation or decommissioning,*
7) *Minimize the generation and volume of radioactive waste both during operation and during decommissioning (by minimizing the volume of components and structures that become contaminated during plant operation)*
8) *Develop a conceptual site model (based on site characterization and facility design and construction) which will aid in the understanding of the interface with environmental systems and the features that will control the movement of contamination in the environment,*
9) *Evaluate the final site configuration after construction to assist in preventing the migration of radio-nuclides offsite via unmonitored pathways,*

Refer to "Impact on DCD" section below for the DCD Tier 2 mark-up and changes.

In addition to the above-mentioned DCD Chapter 12 changes, reference document is also updated. NEI 07-08, Revision 3 is referenced in Subsection 12.1. Since this NEI report is approved by the NRC, DCD Chapter 12 is revised to reference the approved version of NEI 07-08A. Also, NEI 08-08A "Generic FSAR Template Guidance for Life Cycle Minimization of Contamination" is added as a reference in section 12.1.5 of the DCD.

Impact on DCD

Please refer to Attachments A1 and A2 to this RAI response for the DCD mark-up.

Impact on COLA

A COL Applicant action item 12.3(10) has been added to the DCD requiring the COL Applicant to address the site-specific, operational, post-construction objectives, and conceptual site model guidance of Regulatory Guide 4.21. Therefore, the COL applicant will add a new subsection addressing the site-specific design features, operational, post-construction objectives, and conceptual site model guidance of Regulatory Guide 4.21.

Impact on PRA

There is no impact on the PRA

This completes MHI's response to the NRC's question.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

7/30/2010

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 578-4483 REVISION 2**
SRP SECTION:	**12.03-12.04 – Radiation Protection Design Features**
APPLICATION SECTION:	**12.3**
DATE OF RAI ISSUE:	**4/27/2010**

QUESTION NO. : 12.03-12.04-38

20.1406, ISG-006, RG-4.21, SRP 12.03-12.04

RAI-91-1496 Question 12.03-12.04-2 Supplemental Question
In the original RAI the applicant was asked to describe specific examples of the design and operational considerations which demonstrate compliance with 10CFR20.1406. The applicant was asked to identify and add these considerations to the appropriate DCD sections in the USAPWR DCD and provide the basis for these considerations.

The staff had requested that the applicant include in the response how the design and operational objectives, accomplished the following objectives:

- Minimize leaks and spills and provide containment in areas where such events may occur,
- Provide for leak detection capability to provide prompt detection of leakage for any structure, system, or component which has the potential for leakage,
- Use leak detection instrumentation capable of detecting minor leaks in areas where it is difficult or impossible to conduct regular inspections (such as for spent fuel pools, tanks that are in contact with the ground, and buried, embedded, or subterranean piping) to avoid release of contamination from undetected leaks.

US-APWR FSAR Tier 2 Revision 2, Table 11.1-9 "Realistic Source Terms" states that a typical activity of tritium in the secondary side water and steam is 0.001µCi/g. In addition to the tritium activity, the stated activity of the Steam Generator blowdown and main steam system (MSS) exceeds 3 E-6 µCi/g and 1.5E-8 µCi/g, respectively, even when iodines and noble gases are excluded. Contrary to the information provided in FSAR Table 11.1-9, the applicant noted in their response that the Main Condenser is not expected to be a significant source of contamination within PWRs in general and the US APWR in particular.

The applicant was asked to describe design features for any other plant systems which may generate radioactive waste or could result in the contamination of non-radioactive systems. Based on the realistic source term information provided by the applicant, the Condensate and Feedwater System, Main Steam Supply System (MSS), Steam Generator Blowdown System (SGBDS), Auxiliary Steam Supply System (ASSS) and portions of the Main Condensers are examples of systems that are expected to contain low but detectable concentrations of radioactive material.

The information provided in the applicant's RAI-91-1496 Question 12.03-12.04-2 response, does not address environmental contamination at the current generation PWR plants due to undetected leakage and subsequent release of tritium or other isotopes as a result of corrosion of buried piping from these types of systems. For example:

1. The applicant's response did not fully address the Condensate Storage and Transfer System, specifically the underground condensate transfer line from the hotwell to the condensate storage tank (CST), depicted on USAPWR FSAR Revision 2 Tier 2 Figure 12.3-1 as located in the Transformer Yard. As described in the USAPWR DCD Tier 2, Revision 2, section 10.4.1.2.1 "System Operation" states that during normal power operation, the hotwell level controller provides automatic makeup or rejection of condensate to maintain a normal level in the condenser hotwells. On low level, the makeup control valves open and admit condensate to the hotwell from the condensate storage tank. On high-water level, the condensate reject control valves open to divert water from the condensate pump discharge to the condensate storage tank.

2. The applicant's response did not fully address the ASSS, especially, any buried condensate piping. As noted in DCD section 10.4.11 "Auxiliary Steam Supply System", the ASSS is supplied by MSS during operation of the main steam system, or by the auxiliary boiler when main steam is not available. Condensed water from these components is collected in the auxiliary steam drain tank and then is transferred to the condenser during normal operation or to the auxiliary boiler when the main steam is not available. The Condensate Storage Facility supplies make up feed water to the auxiliary boiler. The auxiliary steam system supplies the following components:

    - Boric acid evaporator
    - Boric Acid Batching Tank
    - Non safety-related HVAC equipment
    - Steam for the condensate system deaerator for condensate heating during start up
    - Turbine gland seal
    - Deaerator heating
    - Pre-operational cleanup of the condensate and feedwater system.
    - Steam for maintaining pressure in the condensate system deaerator after a turbine trip.

3. The applicant's response did not fully address the Steam Generator Blowdown System, especially, any buried piping, including the resin transfer lines, and blowdown water discharge line connections to off site disposal systems. The Steam Generator Blowdown System has the following attributes noted in the DCD:

    10.4.8.2.2 "System Operation" In the case of primary to secondary Steam Generator (SG) tube leakage within the allowable tube leak rate, as specified in the plant technical specifications, blowdown water continues to be purified with SG blowdown demineralizers to remove the radioactivity entering from leaking SG tube(s). When the SG tube leak exceeds the allowable limits, the SG blowdown lines are automatically isolated upstream of the SG blowdown demineralizers by the SG blowdown return water radiation monitor high signal.

US-APWR FSAR Tier 2 Revision 2, Chapter 12, identifies some US-APWR general design features that would minimize the contamination of the facility and environment and would minimize the generation of radioactive waste. However, this information does not address design features that are unique to system designs or their locations in the plant warranting more technical details, such as the auxiliary steam and condensate systems. Current plant experience has demonstrated that normal PWR plant operation will likely result in detectable levels of tritium as well as other radionuclides in systems like the Main Steam, Condensate and Auxiliary Steam and Auxiliary Boilers systems.

Supplemental Question:
Please revise and update the US-APWR FSAR Tier 2 to describe the design features for leakage prevention and early leakage detection that will be implemented for the condensate, steam generator blowdown, auxiliary steam and similar systems. If design features for leakage prevention are not applied, please describe how the combined construction and operating license applicants will address these

systems. Include the information in the DCD and provide a markup of the text and appropriate revised diagrams/maps in your response.

ANSWER:

1. The Condensate Polishing System (CPS) is located on the 2nd floor inside the T/B. The CPS is designed to remove dissolved ionic solids and impurities from the condensate and assists in the removal of suspended corrosion products. The condensate still contains a small amount of tritium due to tritium diffusion in the SG tubes and primary to secondary leakage. In the event of valve and piping leaks inside the building, the leakage is collected in the T/B sumps for analysis. If it is determined to be contaminated, the leakage is forwarded to the LWMS for processing.

   The CPS cleans up the entire condensate inventory in the hotwell before plant startup and is usually bypassed during normal operation. The condensate after treatment is returned to the hotwell or sent to the condensate storage tank (CST) to maintain liquid level in the hotwell. The CST is located in the yard and is connected to the hotwell via a condensate transfer pipe which transfers condensate from the hotwell at high liquid level, and transfers makeup at low hotwell liquid level. As a result of industry lessons learned on underground piping, the condensate transfer pipe is a single wall welded stainless steel piping in a coated trench with removable covers. This design is supplemented by operational programs which includes periodic (once every two years) hydrostatic or pressure testing of pipe segments, instrument calibration, and when required, visual inspection and maintenance of piping, trench and instrument integrity. The trench is designed to equip with removable but sealed covers to minimize infiltration of precipitation, and to provide the capability to maintain, repair, and replace of piping and trench when needed. The trench also has inspection manholes with drain collection basin and liquid level switch. When fluid is detected, the instrument initiates alarms for operator actions that include inspection, sampling and analysis to determine the characteristics and source of the fluid, and extraction of the fluid for treatment and disposal. The provisions of inspection manholes, leak detection instruments, and testing of the piping, minimize the need to remove the trench covers to maintain the integrity of the seals. Sections of the manholes are removed only when leakage is detected or when trench repair is needed. The concrete trench has an epoxy coating to facilitate drainage and a barrier for unintended release. This design approach provides maximum accessibility to inspect and test the pipe and the trench, and prompt detection of leak for operator action. Thus this design approach is compliant with the guidance of RG 4.21. COL Item 12.3(10) is included to provide flexibility to the COL Applicant to implement the site-specific design features and operational programs of the condensate piping to prevent and/or minimize leakage. Refer to "Impact on DCD" section below for the DCD Tier 2 markup.

2. The ASSS supplies steam for plant system heating when main steam is not available. The auxiliary boiler takes condensate makeup from the auxiliary steam drain tank inside the A/B, or demineralized water from the demineralized water storage tank in the yard. The auxiliary boiler is located in the yard near the T/B. The condensate piping from the ASSS drain tank is a single-walled stainless steel pipe run above ground in pipe chases from the A/B to the T/B, and is then connected to double-walled welded stainless steel piping through the T/B wall penetration to the auxiliary boiler. Since this is not a high traffic area, this segment of pipe is run above ground and is slightly sloped so that any leakage is collected in the outer pipe and drained to the auxiliary boiler building. At the auxiliary boiler building end, a leak detection instrument is provided to monitor leak. A drain pipe is provided to direct any drains to the building sump. The steam piping is jacketed with insulation and heat protection. This design is supplemented by operational programs which includes periodic (once every two years) hydrostatic or pressure testing of pipe segments, instrument calibration, and when required, visual inspection and maintenance of piping, trench and instrument integrity. Also, this design approach prevents unintended leakage and provides accessibility, and is compliant with the guidance of RG 4.21. COL Item 12.3(10) is included to provide flexibility to the COL Applicant to implement the site-specific design features

of the ASSS to minimize leakage. Refer to "Impact on DCD" section below for the DCD Tier 2 section 10.4.11.2.1 markup.

3. The US-APWR SGBDS employs two subsystems: one for normal operation (described in DCD Subsection 10.4.8); and another for startup operations (site specific design). During normal operation, the SG blowdown Flash Tank located inside the T/B is used. The blowdown water is flashed, cooled, and is then routed to the SGBD ion exchange columns for treatment. The treated blowdown water is sent back to the condenser hotwell for recycle and is not intended for release. The piping in the T/B, and between the T/B, PSB, R/B, and A/B, is routed above ground inside the buildings. There is no buried piping. The piping for the spent resin transfer lines that transfers spent resin to the spent resin storage tanks is also run above ground in pipe chases, and is not buried. In the event that the blowdown water is contaminated, the water, downstream of the SGBD mixed bed demineralizers, is routed to the Waste Holdup Tank (WHT) for treatment and release through the LWMS release piping. During normal operation, the blowdown water is cooled and treated in the mixed bed demineralizers and is totally recycled. The treated water is not sent to the Waste Water System (WWS) for release as indicated in Figure 10.4.8-2. The lines after manual angle valves are connected to WWS via the startup SGBD subsystem as site specific system. This subsystem will be composed of site specific components to cool and release the blowdown to WWS.

   During plant restart after refueling operation, steam generator blowdown is directly flashed into the Startup SG Blowdown Flash Tank located inside the Startup SGBDS Facility located outside the T/B. The Startup SG blowdown Flash Tank is equipped with its own startup SG blowdown heat exchanger to lower the temperature of the blowdown for release to the WWS. The condensate exits the bottom of the Startup SG blowdown Flash Tank, is cooled by the Startup SG blowdown heat exchanger, and its contamination level is monitored by a radiation element designed to detect gross gamma. At this point, the piping is split into a condensate return line and a discharge line. When a pre-determined setpoint is reached, the radiation monitor initiates signals to alarm in the MCR for operator action, closes the isolation valve (located on the discharge line to WWS) and open the return line valve to forward the condensate to the WHT in the Liquid Waste Management System for further processing and release.

   The blowdown condensate return line from the Startup SGBD Facility to the WHT in the A/B is a double-walled pipe designed to run above ground and is kept as short as practical before it penetrates the building wall back to the T/B. The piping is changed to a single-walled pipe and is run in the pipe chases inside the T/B to the WHT in the A/B to minimize pipe leakage to the outside. Leakages inside the T/B and A/B are collected in sumps for processing. The short piping with insulation jackets located outside is inspected periodically to check for leakage.

   During Steam Generator (SG) restart operation, which is anticipated to be a few hours for each SG restart, the condensate is cooled and is directed to a WWS for release. This discharge line consists of the following piping segments:

   1. Single-walled stainless steel pipe from the heat exchanger up to and including the radiation monitor and the valves inside the Startup SGBDS Facility. This line section includes the condensate return line and the discharge piping;
   2. The discharge piping segment, including the portion through the wall penetrations, is then connected to double-walled stainless steel piping from the Startup SGBDS Facility through the T/B wall penetration;
   3. Once inside the T/B, the discharge piping is connected to single-walled stainless steel pipe and is routed in pipe chases;
   4. From the pipe chase, the discharge pipe exits the T/B penetration and is routed as a single-walled stainless steel piping in a concrete trench from the T/B to the transiting manhole downstream of the Condensate Storage Tanks (CST). This portion of the piping is in the same concrete trench for the condensate transfer piping to the CST. The concrete trench has an epoxy coating to facilitate drainage and a barrier for unintended release. Using single-wall stainless steel pipe in the trench facilitates

additional radial cooling of the fluid and enables the use of HDPE piping for underground burial;

5. From the transition manhole the discharge piping is connected to a buried double-walled HDPE piping to an existing WWS discharge. Transition manhole is constructed near the plant pavement boundary. HDPE pipe has good corrosion resistant property in the soil environment.

Additional manholes are provided for testing and inspection for the buried piping. Each manhole is equipped with drain collection basins and leak detection instruments. This design approach minimizes unintended releases and provides accessibility to facilitate periodic (hydrostatic or pressure) testing and visual inspection, and to maintain pipe integrity. This design approach is also compliant with RG 4.21. COL Item 12.3(10) is included to provide flexibility to the COL Applicant to implement the site-specific design features of the Startup SGBDS to minimize leakage.

Other piping associated with the SGBD operation, and those of other systems that carry radioactively contaminated liquid, or that may become contaminated, are designed with welded connections to reduce leakage through flanged or screwed connections. Piping materials are compatible with the transferred fluid to preserve water quality and minimize pipe corrosion for the life of the plant.

In addition to the above, information on the Primary Makeup Water (PMW) Tank piping will also be added to the DCD. The piping to and from the PMW Tank is single-walled stainless steel piping designed to run aboveground and penetrates the building wall directly into the tank. This piping is mostly inside the A/B in pipe chases. For piping between buildings, penetration sleeves are provided to collect and direct any leakages back into the building for further processing. The PMW Tanks and the Refueling Water Storage Auxiliary Tank are housed in a tank enclosure which protects the tanks and the piping from the environment. Similar piping is provided for the PMW Tanks carrying recycle water back to the A/B. This design is supplemented by operational programs which includes periodic (once every two years) hydrostatic or pressure testing of pipe segments, and visual inspections to maintain piping integrity. This design approach is also compliant with RG 4.21. COL Item 12.3(10) is included to provide flexibility to the COL Applicant to implement the site-specific design features of the PMWTs and the RWSAT to prevent and/or minimize leakage. Refer to "Impact on DCD" section below for the DCD Tier 2 section 9.2.6.2.6 markup.

Also, the general statement for periodic inspection of the Condensate Storage Facility (CSF) tanks and piping is added to the DCD. Refer to the "Impact on DCD" section below for the markup.

Impact on DCD

1. Add the following as second paragraph to DCD section 9.2.6.2.4 "Condensate Storage Tank",

   "The transfer piping running between the CST and the hotwell is single-walled welded stainless steel piping in a coated trench with removable but sealed covers. This design is supplemented by periodic hydrostatic or pressure testing of pipe segments, instrument calibration, and when required, visual inspection and maintenance of piping, trench and instrument integrity, in compliance with the guidance of RG 4.21 and industry operating experience. Design and system features addressing RG 4.21 are captured in Section 12.3.1.3 of the DCD."

2. Add the following to the third paragraph to DCD Subsection 10.4.11.2.1 "General Description", after the third bullet (non safety-related HVAC equipment),

   "The ASSS supplies steam for plant system heating when main steam is not available. The auxiliary boiler takes condensate makeup from the auxiliary steam drain tank inside the A/B, or from the condensate storage tank (CST) in the yard. The auxiliary boiler is

located in the yard near the plant area. The condensate piping from the ASSS drain tank is a single-walled stainless steel pipe run above ground in pipe chases from the A/B to the T/B, and is then connected to double-walled welded stainless steel piping through the T/B wall penetration to the auxiliary boiler. Since this is not a high traffic area, this segment of pipe is run above ground and is slightly sloped so that any leakage is collected in the outer pipe and drained to the auxiliary boiler building. At the auxiliary boiler building end, a leak detection instrument is provided to monitor leak. A drain pipe is provided to direct any drains to the building sump. The steam piping is jacketed with insulation and heat protection. This design is supplemented by operational programs which includes periodic hydrostatic or pressure testing of pipe segments, instrument calibration, and when required, visual inspection and maintenance of piping, trench and instrument integrity.

A discussion of the radiological aspects of the system leakage is contained in DCD Section 11.1. Design and system features addressing RG 4.21 are captured in Section 12.3.1.3 of the DCD."

3. No changes to the DCD per response to item 3 above.

However, as indicated in the response above the following paragraph is added to the DCD:

Add the following as the second paragraph to DCD Tier 2, Section 9.2.6.2.6 "Primary Makeup Water Tanks"

"The piping to and from the PMW Tank is single-walled stainless steel piping designed to run aboveground and penetrates the building wall directly into the tank. This piping is mostly inside the A/B in pipe chases. For piping between buildings, penetration sleeves are provided to collect and direct any leakages back into the building for further processing. The piping may require heat tracing to protect against freezing. The PMWTs employ non-leakage type valves such as diaphragm-type valves, or leak control valves with graphite packing for handling radioactive fluid, or leak-off connection is provided to prevent leakage to environment. Similar piping is provided for the PMW Tanks carrying recycle water back to the A/B. This design is supplemented by operational programs which includes periodic hydrostatic or pressure testing of pipe segments, and visual inspections to maintain piping integrity. A discussion on minimizing radioactive contamination of the system is contained in DCD Section 12.3.1.3."

Add the following sentence towards the end of paragraph in DCD Tier 2 section 9.2.6.4,

"CSF tanks including PMWTs and CST and their associated piping are periodically tested / inspected for leakages."

Impact on COLA

There is no impact on the COLA

Impact on PRA

There is no impact on the PRA

This completes MHI's response to the NRC's question.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

7/30/2010

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 578-4483 REVISION 2**
SRP SECTION:	**12.03-12.04 – Radiation Protection Design Features**
APPLICATION SECTION:	**12.3**
DATE OF RAI ISSUE:	**4/27/2010**

QUESTION NO. : 12.03-12.04-39

20.1406, ISG-006, RG-4.21, SRP 12.03-12.04

USAPWR RAI 91-1496, Question 12.03-12.04-2, 10CFR20.1406 Supplemental Question

US-APWR DCD RAI 91-1496 Question, 12.03-12.04-2 asked the applicant to address how they will comply with the requirements of 10 CFR 20.1406. The question specifically asked the applicants to address radioactivity in systems such as the Main Condenser and any other plant systems which may generate radioactive waste or could result in the contamination of non-radioactive systems.

US-APWR FSAR Tier 2, Revision 2, Section 11.1.1.3 "Tritium" notes that the activity of H-3 in the secondary side water and steam is entirely controlled by the loss of water from the reactor coolant system through primary-to-secondary leakage. Section 10.1.2 "Protective Features" paragraph "Radioactivity Protection" states that under normal operating conditions, there are no radioactive contaminants of operational concern present in the steam and power conversion system; however, it is possible for the system to become contaminated through steam generator tube leakage. Section 10.3.3 "Safety Evaluation" notes that radioactive contamination of the Main Steam System (MSS) can occur by a primary side to secondary side leak in the Steam Generator (SG). Under normal operating conditions, there are no significant amounts of radioactivity in the Main Steam System (MSS). The applicant noted in their response to RAI-91-1496, Question 12.03-12.04-2 that the Main Condenser is not expected to be a significant source of contamination within PWRs in general and the US-APWR in particular.

However, these statements are not consistent with DCD Tier 2, Revision 2 Section 11.1.1.3 which states that typical activity of tritium in the secondary side water and steam is 0.001μCi/g. As noted in Table 11.1-4 "Parameters Used to Calculate Secondary Coolant Activity" which is based on 150 gallons per day of primary to secondary leakage. Also, information from currently operating plants shows that in the absence of primary to secondary leakage, tritium activity is still present in secondary side water and steam at concentrations determined by the rate of hydrogen perfusion through the SG U-Tubes, the amount of secondary side reuse and the secondary side make up rate. This information shows that 2,000 to 100,000 pico Curies/liter of tritium will be present in the secondary side steam/condensate fluid. As the US-APWR design employs full reuse of SG blowdown water and anticipated secondary side make up rates are expected to be low, the secondary side tritium concentration is likely to be higher than in many of the current generation of PWR plants. Experience from currently operating plants, shows that leakage from these types of systems will represent an operational concern, for plants subject to the requirements of 10 CFR 20.1406.

Please revise and update US-APWR FSAR Tier 2, Revision 2 Chapter 10 and Section 11.1.1.3 to incorporate a description of the expected radioactivity in secondary plant fluid systems in the absence of identified primary to secondary leakage consistent with the source term provide in Section 11.1.1.3 and currently operating plant data.

ANSWER:

DCD Revision 2 Subsection 11.1.1.3 discusses typical activity of tritiated water (tritium) in the secondary side water and steam as being 0.001 micro-Curies per gram (equivalent to 1E+ 6 pCi/L [pico-Curies per liter]). This activity is based on a primary-to-secondary leakage of 75 lb/day, stipulated in ANSI/ANS 18.1-1999, Table 6). This estimate is also consistent with ANSI/ANS 18.1 concentration estimate, which is based on long term operation and includes "moderate (as differentiated from total)" condensate recycle. This calculated activity is higher than the typical activities of 2E+3 to 1E+ 5 pCi/L observed in industry secondary side steam/condensate fluid without primary-to-secondary leakage. Hence, this calculated activity is representative of normal operation with a small amount of primary-to-secondary leakage (75 lb/day).

The calculated tritium activity, based on a leakage rate of 150 gallons per day (gpd) (equivalent to 890 lb/day, based on average core temperature), with all condensate being recycled, could approach a significant fraction of reactor coolant concentrations of 3.5 micro-Curies per gram.

A footnote will be added to the headers of Table 11.1-9 to clarify that the SG Water and Steam Activities are based on a primary-to-secondary leakage rate of 75 lb/day and resulting secondary coolant activity as described in ANSI/ANS-18.1-1999.

DCD Section 10.1.2, subsection "Radioactivity Protection" provides a discussion of the potential radioactive contamination in the steam and power conversion system. This section points to a discussion of the radiological aspects of primary-to-secondary system leakage and conditions for operation contained in Chapter 11. Hence there is no change required for this section.

Refer to "Impact on DCD" section below for the changes to the DCD Tier 2 per this RAI response.

Impact on DCD

DCD Section 10.3.3 "Safety Evaluation", fifth bullet is revised as follows:

"Radioactive contamination of the MSS can occur by a primary side to secondary side leak in the SG. ~~Under normal operating conditions, there are no significant amount of radioactivity in the MSS.~~ **The main steam can become contaminated due to tritium diffusion through SG tubes even without primary-to-secondary leakage. A discussion of the radiological aspects of primary-to-secondary system leakage and conditions for operation is contained in Chapter 11.** Additionally, the MSIVs provide controls for reducing releases by isolating the affected main steam line following a steam generator tube rupture (SGTR). In-line radiation monitors on each steam line, condenser vacuum pump exhaust line radiation monitor, GSS exhaust fan discharge line radiation monitor and the SG blowdown line radiation monitor facilitate leak detection."

DCD Section 10.4.8.3 "Safety Evaluation", third bullet is revised as follows:

"Radioactive contamination of the SGBDS can occur by a primary to secondary leakage in the steam generator. ~~Under normal operating conditions, there is no significant amount of radioactivity in the steam generator blowdown.~~ **The SGBDS can become contaminated due to tritium diffusion through SG tubes even without primary-to-secondary leakage. A discussion of the radiological aspects of primary-to-secondary system leakage and conditions for operation is contained in Chapter 11.** The isolation valve(s) in each blowdown line provides controls for reducing releases by isolating the affected steam generator blowdown line following a steam generator tube rupture. An inline radiation monitor on the common line from the steam generator blowdown sample lines, facilitate leak detection."

Add the following paragraph after the last sentence in DCD Subsection 11.1.1.3,

"This activity is calculated based on a primary-to-secondary leakage rate of 75 lb/day with a moderate amount of condensate recycle. At higher primary-to-secondary leakage rate, up to and including 150 gallons per day, and with full recycle of condensate, tritium concentration is progressively higher, approaching reactor coolant concentration."

Add a footnote to the headers of Table 11.1-9 as follows,

"SG Water and Steam Activities are based on a primary-to-secondary leakage rate of 75 lb/day and resulting secondary coolant activity as described in ANSI/ANS-18.1-1999."

Impact on COLA

There is no impact on the COLA

Impact on PRA

There is no impact on the PRA

This completes MHI's response to the NRC's question.

ATTACHMENT A1 to RAI 578-4483
Response 12.03-12.04-37

12.0 RADIATION PROTECTION

12.1 Ensuring that Occupational Radiation Exposures are As Low As Reasonably Achievable

US-APWR is to keep all radiation exposure of personnel within limits defined by Title 10, Code of Federal Regulations, Part 20 (Reference 12.1-1). Administrative procedures and practice in US-APWR related to maintaining radiation exposure of personnel as low as reasonably achievable (ALARA) are described below, referring to NEI 07-08A (Reference 12.1-2) submitted in August 2007 to the U.S. Nuclear Regulatory Commission (NRC).

12.1.1 Policy Considerations

The facility design, administrative programs and procedures ensure that occupational radiation exposure to personnel is kept ALARA. The organization of responsibilities for the design and the operation of the US-APWR are intended to achieve ALARA occupational radiation exposures.

12.1.1.1 Design Policies

The US-APWR is designed to take into account the ALARA philosophy to reduce occupational radiation exposure during normal operation and accident conditions. The ALARA philosophy was applied during the initial design of the plant and implemented through internal design reviews. The design has been reviewed in detail for ALARA considerations, and will be reviewed, updated, and modified, as necessary, during the detail design phase, and as experience is obtained from operating plants. Nuclear engineers with extensive experience in ALARA design and operation reviewed the plant design, integrated the layout, shielding, ventilation, and monitoring instrument designs with the traffic control, security, access control, and health physics aspects of the design and operation to ensure that the overall design is conducive to maintaining exposures ALARA.

All pipe routing containing radioactive fluids is reviewed as part of the engineering design effort. This ensures that lines expected to contain significant radiation sources are adequately shielded and properly routed to minimize exposure to personnel.

Lessons learned from operating plants are continuously integrated into the design of the US-APWR.

The manager of the section responsible for radiation protection engineering requires the ALARA design. The managers of related design sections develop the design for ensuring ALARA, according to the requirement from the manager of the section that is responsible to radiation protection engineering.

12.1.5 References

12.1-1 "Standards for Protection Against Radiation," Energy. Title 10, Code of Federal Regulations, Part 20, U.S. Nuclear Regulatory Commission, Washington, DC, May 1991.

12.1-2 Generic FSAR Template Guidance for Ensuring That Occupational Radiation Exposures Are As Low As Is Reasonably Achievable (ALARA). NEI Technical Report 07-08**A**, Revision **0**, **Oct. 2009**.

12.1-3 Qualification and Training of Personnel for Nuclear Power Plants. RG 1.8, Rev. 3, U.S. Nuclear Regulatory Commission, Washington, DC, May 2000.

12.1-4 Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable. RG 8.8, Rev. 3, U.S. Nuclear Regulatory Commission, Washington, DC, June 1978.

12.1-5 Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as Is Reasonably Achievable. RG 8.10, Rev. 1-R, U.S. Nuclear Regulatory Commission, Washington, DC, May 1977.

12.1-6 "Minimization of Contamination." Energy. Title 10 Code of Federal Regulations, Part 20.1406, U.S. Nuclear Regulatory Commission, Washington, DC.

12.1-7 Guide for Administrative Practices in Radiation Monitoring. RG 8.2, Rev.0, U.S. Nuclear Regulatory Commission, Washington, DC, February 1973.

12.1-8 Direct-Reading and Indirect-Reading Pocket Dosimeters. RG 8.4, Rev.0, U.S. Nuclear Regulatory Commission, Washington, DC, February 1973.

12.1-9 Standard Test Procedure for Geiger-Müller Counters. RG 8.6, Rev.0, U.S. Nuclear Regulatory Commission, Washington, DC, May 1973.

12.1-10 Instructions for Recording and Reporting Occupational Radiation Exposure Data. RG 8.7, Rev. 2, U.S. Nuclear Regulatory Commission, Washington, DC, November 2005.

12.1-11 Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program RG 8.9, Rev.1, U.S. Nuclear Regulatory Commission, Washington, DC, July 1993.

12.1-12 Instruction Concerning Prenatal Radiation Exposure. RG 8.13, Rev.3, U.S. Nuclear Regulatory Commission, Washington, DC, Jun80e 1999.

12.1-13 Acceptable Programs for Respiratory Protection. RG 8.15, Rev.1, U.S. Nuclear Regulatory Commission, Washington, DC, October 1999.

12.1-14 Deleted.

12.1-15 Air Sampling in the Workplace. RG 8.25, Rev.1, U.S. Nuclear Regulatory Commission, Washington, DC, June 1992.

12.1-16 Deleted.

12.1-17 Radiation Protection Training for Personnel at Light-Water-Cooled Nuclear Power Plants. RG 8.27, Rev.0, U.S. Nuclear Regulatory Commission, Washington, DC, March 1981.

12.1-18 Audible-Alarm Dosimeters. RG 8.28, Rev.0, U.S. Nuclear Regulatory Commission, Washington, DC, August 1981.

12.1-19 Instruction Concerning Risks from Occupational Radiation Exposure. RG 8.29, Rev.1, U.S. Nuclear Regulatory Commission, Washington, DC, February 1996.

12.1-20 Deleted.

12.1-21 Monitoring Criteria and Methods To Calculate Occupational Radiation Doses. RG 8.34, Rev.0, U.S. Nuclear Regulatory Commission, Washington, DC, July 1992.

12.1-22 Planned Special Exposures. RG 8.35, Rev.0, U.S. Nuclear Regulatory Commission, Washington, DC, June 1992.

12.1-23 Radiation Dose to the Embryo/Fetus. RG 8.36, Rev.0, U.S. Nuclear Regulatory Commission, Washington, DC, July 1992.

12.1-24 Control of Access to High and Very High Radiation Areas of Nuclear Plants. RG 8.38, Rev.1, U.S. Nuclear Regulatory Commission, Washington, DC, May 2006.

12.1-25 Generic FSAR Template Guidance for Radiation Protection Program Description. NEI Technical Report 07-03A Revision 0, May. 2009.

12.1-26 Quality Assurance Program Requirements (Operation). RG 1.33, Rev.2, U.S. Nuclear Regulatory Commission, Washington, DC, February 1978.

12.1-27 Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning. RG 4.21, Rev.0, U.S. Nuclear ~~Regulatory Commision~~ **Regulatory Commission**, Washington, DC, June 2008.

12.1-28 Generic FSAR Template Guidance for Life Cycle Minimization of Contamination" NEI Technical Report 08-08A, Revision 0, October 2009

12.3.1.3 Minimization of Contamination and Radioactive Waste Generation

This section describes the US-APWR design features and operational programs to address the requirements of 10 CFR 20.1406 (Reference 12.3-29) and RG 4.21 (Reference 12.3-30). These design features, working in conjunction with operational programs, minimize environmental contamination and the generation of radioactive waste, maintain occupational doses ALARA, and facilitate the eventual decommissioning of the facility.

12.3.1.3.1 Design Considerations

The requirements of Regulatory Position C.1 through C.4 of Regulatory Guide 4.21 (Reference 12.3-30) can be met by addressing the following design objectives.

- Objective 1 – Minimize leaks and spills and provide containment in areas where such events might occur.
- Objective 2 – Provide adequate leak detection capability to provide prompt detection of leakage for any structure, system or component that has the potential for leakage.
- Objective 3 – Use leak detection methods (e.g., instrumentation, automated samplers) capable of early detection of leaks in areas where it is difficult or impossible to conduct regular inspections (such as for spent fuel pools, tanks that are in contact with the ground, and buried, embedded, or subterranean piping) to avoid release of contamination of environment.
- Objective 4 – Reduce the need to decontaminate equipment and structures by decreasing the probability of any released, reducing any amounts released, and decreasing the spread of the contaminant from the source.
- Objective 5 – Facilitate decommissioning by (1) minimizing embedded and buried piping, and (2) designing the facility to facilitate the removal of any equipment and/or components that may require removal and/or replacement during facility operation or decommissioning.
- Objective 6 – Minimize the generation and volume of radioactive waste both during operation and during decommissioning (by minimizing the volume of components and structures that become contaminated during plant operation).

Subsection 12.1.2.1 describes the US-APWR design features used to maintain personnel exposures ALARA and reduce the plant source term. Some of these design features also address several RG 4.21 (Reference 12.3-30) design objectives and ultimately help minimize contamination, reduce the generation of radioactive waste, and facilitate decommissioning. Additionally, many of the US-APWR structures, systems, and components (SSCs) incorporate specific design features designed to meet the requirements of 10 CFR 20.1406 (Reference 12.3-29) and RG 4.21 (Reference 12.3-30). These design features are captured in Table 12.3-8 "Design Features for Minimizing Contamination and Generation of Radioactive Waste". The Table also provides cross-references to the applicable DCD subsections that describe how these SSCs address these design objectives to provide compliance with the requirements of 10 CFR 20.1406 (Reference 12.3-29).

12.3.1.3.2 Operational/Programmatic Considerations

Operational programs and procedures that address the requirements of 10 CFR 20.1406 (Reference 12.3-29) are necessary adjuncts to the design features discussed in the Subsection 12.3.1.3.1. The operational and post-construction requirements of Regulatory Position C.1 through C.4 of RG 4.21 (Reference 12.3-30) can be met by addressing the following operational/programmatic objectives.

- Periodically review operational practices to ensure that, operating procedures are revised to reflect the installation of new or modified equipment, personnel qualification and training are kept current, and facility personnel are following the operational procedures.

- Facilitate decommissioning by maintenance of records relating to facility design and construction, facility design changes, site conditions before and after construction, onsite waste disposal and contamination and results of radiological surveys.

- Develop a conceptual site model (based on site characterization and facility design and construction) which will aid in the understanding of the interface with environmental systems and the features that will control the movement of contamination in the environment.

- Evaluate the final site configuration after construction to assist in preventing the migration of radionuclides offsite via unmonitored pathways.

- Establish and perform an onsite contamination monitoring program for the potential pathways from the release sources to the receptor points.

COL action items 12.1(6), 12.1(7), 12.1(8) and 12.3(10) are incorporated to capture the above operational and programmatic objectives.

The alarm setpoint is determined by providing a margin relative to the normal radiation levels, as the main purpose of the alarm is to detect any abnormal situation.

The airborne radioactivity monitoring system is 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.

12.3.5 Dose Assessment

Refer to Section 12.4 for the discussion of the dose assessment.

12.3.6 Combined License Information

COL 12.3 (1)	*The COL Applicant is responsible for the use of portable instruments, and the associated training and procedures, to accurately determine the airborne iodine concentration in areas within the facility where plant personnel may be present during an accident, in accordance with the requirements of 10 CFR 50.34(f)(2)(xxvii) and the criteria in Item III.D.3.3 of NUREG-0737.*
COL 12.3 (2)	*Deleted.*
COL 12.3 (3)	*Deleted.*
COL 12.3 (4)	*The COL Applicant is to provide the site radiation zones that is shown on the site-specific plant arrangement plan.*
COL 12.3 (5)	*The COL Applicant is to discuss the administrative control of the fuel transfer tube inspection and the access control of the area near the seismic gap below the fuel transfer tube.*
COL 12.3 (6)	*If the COL Applicant adopts the Mobile Liquid Waste Processing System, the COL Applicant is to provide information about the radiation protection aspects of the system and to indicate how the system is consistent with the guidance in SRP Section 12.3-12.4, RG 1.206 C.I.12.3.2 and RG 1.69.*
COL 12.3 (7)	*If the COL applicant adopts the Mobile Liquid Waste Processing System, the Col applicant is to provide information about prevention and detection of contamination of the environment and minimization of decommissioning costs and to explain how the system meets the requirements of 10 CFR 20.1406 and RG 4.21.*
COL 12.3 (8)	*If the COL Applicant adopts the Mobile Liquid Processing System, the COL Applicant is to confirm the radiation zone(s) where the system is installed in and to revise Figure 12.3-1, if necessary.*
COL 12.3 (9)	*In order to ensure that the B.A. evaporator room does not become a VHRA during the end of cycle, the COL Applicant is to stipulate a need for routine surveillance in the Radiation Protection Program. In the event that the routine surveillance shows an increase in dose level, the COL Applicant must provide an appropriate strategy to sufficiently reduce the dose rate below the criteria for a VHRA.*
COL 12.3 (10)	***The COL Applicant will address the site-specific design features, operational, post-construction objectives, and conceptual site model guidance of Regulatory Guide 4.21.***

12.3-28 IEEE Standards, IEEE 344 -1975, IEEE 308-1974, IEEE 497-2002

12.3-29 "Minimization of Contamination." Energy Title 10 Code of Federal Regulations Part 20.1406, U.S. Nuclear Regulatory Commission, Washington, DC.

12.3-30 Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning. RG 4.21, Rev.0, U.S. Nuclear Regulatory Commission, Washington, DC, June 2008

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 1 of 52)

Fuel Storage and Handling **(Note: The "System Features" column consists of excerpts/summary from the DCD)**

Objective		System Features	DCD Reference
1	Minimize leaks and spills and provide containment in areas where such events may occur.	New and spent fuel storage facilities are located in the fuel handling area of the Reactor Building (R/B) which is designed to meet the Seismic Category I requirements of Regulatory Guide (RG) 1.29.	9.1.1.1 9.1.2.2.2
		The fuel storage and handling area is protected against natural phenomena. The robust concrete walls and ceiling surrounding the fuel storage and handling area are designed to withstand the loads and forces caused by wind, tornados, floods, and external missiles.	9.1.1.1 9.1.2.2.2
		The spent fuel storage pit is constructed of reinforced concrete lined with stainless steel plate. Similarly, the refueling canal, fuel inspection pit, and cask pit are constructed of reinforced concrete lined with stainless steel plate.	9.1.2.2.2
		Penetrations for the drain and makeup lines are located to preclude the draining of the SFP due to a break in a line or failure of a pump to stop. The connection for the SFP pumps' suction is located below normal water level and above the level needed to provide sufficient water for shielding and for cooling of the fuel if the SFPCS is unavailable. This design feature aids in minimizing the leakages and spills (dispersion of water) from the SFP.	9.1.2.2.2

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 2 of 52)

Objective		System Features	DCD Reference
		The SFP is not connected to the equipment drain system to preclude unanticipated drainage.	9.1.2.2.2
		Heating, Ventilation, and Air Conditioning (HVAC) provides ventilation for the fuel handling area to maintain the atmospheric pressure in this area slightly negative with respect to outside the building.	9.1.1.1 9.1.2.1 9.1.2.2.2
2	Provide for adequate leak detection capability to provide prompt detection of leakage for any structure, system, or component which has the potential for leakage.	The SFPCS is designed to collect system leakage; instrumentation is provided to indicate SFP water level.	9.1.3.1
		A fuel pool liner leakage collection system is provided to collect possible leakage from liner plate welds on the pit walls and floor. This system is provided with a leak detection capability and alarm. Leaked water is directed to the R/B drain sump.	9.1.2.1 9.1.2.2.2
		A liquid level transmitter is installed in the SFP to monitor water level. The water level indication, high water level alarm, and low water level alarm are relayed to the MCR. A local alarm is also installed for detection by personnel present in the vicinity of the SFP.	9.1.3.5.4

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 3 of 52)

Objective		System Features	DCD Reference
3	Use leak detection methods (e.g., instrumentation, automated samplers) capable of early detection of leaks in areas where it is difficult or impossible to conduct regular inspections (such as for spent fuel pools, tanks that are in contact with the ground, and buried, embedded, or subterranean piping) to avoid release of contamination of the environment.	A SFP liner leakage collection system is provided to collect possible leakage from liner plate welds on the pit walls and floor. This system is provided with a leak detection capability and alarm. Leaked water is directed to the R/B drain sump.	9.1.2.1
		The spent fuel storage pit and its liner are designed to maintain their structural integrity and remain leak tight under all applicable design loads and load combinations. The walls of the SFP are an integral part of the seismic category I reactor building structure.	9.1.2.2.2
		Any leakage from liner plate welds is detected by opening the valves or caps on patrols conducted weekly. To meet the requirements of 10CFR20.1406, the inside of the spent fuel pit leakage collection pipes are inspected using a device such as a fiberscope approximately every refueling outage. Should materials such as accumulated boric acid residue and minerals be detected, the inside of the pipes are cleaned. The spent fuel pool leakage collection pipes are sized to allow cleaning of blockages as specified in RG 4.21.	
		The SFP water chemistry can be checked at local sample points. If purification is required, a portion of the system flow is diverted through the SFP demineralizer and filter and returned to the pit.	9.1.3.2.2.1
		The SFPCS is designed to collect system leakage. A liner collection system to the R/B sump is provided to collect possible leakage from the SFP liner plate welds on the pit walls and floor. Leakage from the system piping is collected to the R/B sump. A leakage alarm will be installed upstream of the R/B sump for immediate detection of significant leakage levels.	9.1.3.1

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 4 of 52)

Objective		System Features	DCD Reference
4	Reduce the need to decontaminate equipment and structures by decreasing the probability of any release, reducing any amounts released, and decreasing the spread of the contaminant from the source.	The spent fuel pit purification and cooling system (SFPCS) includes the following functions: - Purifies and clarifies the SFP water - Purifies the boric acid water for the refueling water storage pit (RWSP), the reactor cavity, and the refueling water storage auxiliary tank (RWSAT) in conjunction with the Refueling Water System (RWS)	9.1.3 9.1.3.2.2.3

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 5 of 52)

Objective		System Features	DCD Reference
		The SFP water chemistry can be checked at local sample points. If purification is required, a portion of the system flow is diverted through the SFP demineralizer and filter and returned to the pit.	9.1.3.2.2.1
		The SFPCS capability is sufficient to permit the necessary operations that must be conducted in the SFP area. The SFPCS is designed to perform its purification function in accordance with the following additional criteria: - The water purification loop contains a filter vessel with a disposable cartridge filter and a mixed bed demineralizer upstream of the filter. - Local sample lines are provided at the SFP pumps discharge lines and the demineralizer outlet lines. Sampling and analysis of SFP water for gross activity and particulate concentration is conducted when the SFPCS is in continuous operation.	9.1.3.2.2.3
		Contamination spread is prevented by such means as: - Processing of water used for refueling and flushing of steel liners following spent fuel handling - Refueling equipment decontamination after use	

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 6 of 52)

Objective		System Features	DCD Reference
5	Facilitate decommissioning by (1) minimizing embedded and buried piping, and (2) designing the facility to facilitate the removal of any equipment and/or components that may require removal and/or replacement during facility operation or decommissioning.	a) Embedded and buried piping is minimized (leak chases are not considered piping). b) Process equipment items are designed with flushing capability, and to be accessible for maintenance and replacement. These items are small in volume and may be removed and replaced with minimum cutting into small pieces. Large tanks are designed for extended life and will not need to be replaced. If a leak develops, these tanks can be accessed for repairs in place, or can be decontaminated to remain in the cubicles until decommissioning, during which time they can be cut into smaller pieces for disposal.	—
6	Minimize the generation and volume of radioactive waste both during operation and during decommissioning (by minimizing the volume of components and structures that become contaminated during plant operation).	Operational contamination is prevented by such means as: - Processing of water used for refueling and flushing of steel liners following spent fuel handling - Refueling equipment decontamination after use	—

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 7 of 52)

Water Systems **(Note: The "System Features" column consists of excerpts/summary from the DCD)**

Essential Service Water System (ESWS)

The water in the ESWS does not normally contain radioactivity (DCD Section 9.2.1.2.2.6). Radioactive contamination only occurs if the CCWS system is contaminated and then leaks into the ESWS via the CCW HX (DCD Section 9.2.1.3); therefore, the following design features are provided.

Objective		System Features	DCD Reference
1	Minimize leaks and spills and provide containment in areas where such events may occur.	**Design features to minimize leaks**	
		The CCWS is the intermediate loop between the components containing radioactive fluids and the ESWS. This arrangement prevents direct leakage of the radioactive fluid to the environment through the ESWS.	9.2.1.2.1
		The water in the ESWS does not normally contain radioactivity (Section 9.2.1.2.2.6). Radioactive contamination only occurs if the CCWS system is contaminated and then leaks into the ESWS via the CCW HX (Section 9.2.1.3). Radioactive release to the environment through the ESW is therefore minimized.	9.2.1.2.2.6, 9.2.1.3
		The other components (other than the CCW HX) cooled by the ESW are the essential chiller units that do not contain radioactive fluid.	9.2.1.2.3.1

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Objective		System Features	DCD Reference
		Also, the effect of long-term corrosion of the piping is mitigated by adding a corrosion inhibitor. The ESW is periodically sampled and chemicals are added, as required, during normal operation. **Design features to provide containment** Radiation monitors are provided in each discharge line of CCW HX essential service water (ESW) side. The monitors alert the operator if the leaking CCW contains radioactivity so that the operator can isolate the train of the ESW which is connected to the leaking CCW HX. The CCWS is designed to serve as an intermediate system between components containing radioactive fluids, which are cooled by the system, and the ESWS so as to prevent direct leakage of radioactive fluid into the environment through the ESWS.	 9.2.1.2.1 9.2.2.1.2
2	Provide for adequate leak detection capability to provide prompt detection of leakage for any structure, system, or component which has the potential for leakage.	Radiation monitors are provided in each discharge line of CCW HX essential service water (ESW) side. The monitors alert the operator if the leaking CCW contains radioactivity so that the operator can isolate the train of the ESW which is connected to the leaking CCW HX. If, however, radioactive leakage does occur in the CCWS, radiation monitors will alarm in the MCR to enable immediate stoppage of the CCW pump and isolation of the leaking train.	9.2.1.2.1 9.2.1.2.3

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Objective		System Features	DCD Reference
3	Use leak detection methods (e.g., instrumentation, automated samplers) capable of early detection of leaks in areas where it is difficult or impossible to conduct regular inspections (such as for spent fuel pools, tanks that are in contact with the ground, and buried, embedded, or subterranean piping) to avoid release of contamination of the environment.	The leak detection instrumentation is described above, and is included on all four trains of the ESWS.	9.2.1.2.1
4	Reduce the need to decontaminate equipment and structures by decreasing the probability of any release, reducing any amounts released, and decreasing the spread of the contaminant from the source.	The ESWS draws water from the intake basin and returns water to the UHS after passing through the CCW HXs and the essential chiller units. The UHS is the source of water to the intake basin. The essential chiller units do not include the radioactive fluid, and CCWS is the intermediate loop between the reactor auxiliaries and the ESWS. This arrangement minimizes direct leakage of the radioactive fluid from the ESWS to the environment. In addition, radiation monitors are provided in each discharge line of CCW HX essential service water (ESW) side. The monitors alert the operator if the leaking CCW contains radioactivity so that the operator can isolate the leaking train.	9.2.1.2
5	Facilitate decommissioning by (1) minimizing embedded and buried piping, and (2) designing the facility to facilitate the removal of any equipment and/or components that may require removal and/or replacement during facility operation or decommissioning.	a) Underground piping is lined and placed in trenches, with manholes provided for periodic piping inspection. b) Process equipment items are accessible for maintenance and replacement. These items are small in volume and may be removed and replaced with minimum cutting into small pieces.	9.2.1.2.2.5

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 10 of 52)

Objective		System Features	DCD Reference
6	Minimize the generation and volume of radioactive waste both during operation and during decommissioning (by minimizing the volume of components and structures that become contaminated during plant operation).	All of the features listed above minimize the possibility of contamination of components and structures, and therefore minimize the volume of radioactive waste generated during operation and decommissioning.	--

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 11 of 52)

Water Systems (Note: The "System Features" column consists of excerpts/summary from the DCD)

Component Cooling Water System (CCWS)

The CCWS normally does not contain radioactive fluids and is a closed loop system. It is designed to serve as an intermediate system between components containing radioactive fluids, which are cooled by the system, and the ESW so as to prevent direct leakage of radioactive fluid into the environment through the ESW system (DCD Section 9.2.2.1.2). Radioactive contamination of CCWS occurs only if there are leakages from radioactive users into the CCWS.

Objective		System Features	DCD Reference
1	Minimize leaks and spills and provide containment in areas where such events may occur.	The system is designed to assure that leakage of radioactive fluid from the cooled components is held within the plant.	1.2.1.5.4.4
		Design features to minimize leaks	
		Water chemistry control of CCWS is performed by adding chemicals to the CCW surge tank to prevent long term corrosion that may degrade system performance. The CCW in the surge tank is covered with nitrogen gas to maintain water chemistry.	9.2.2.3.4
		Piping joints and connections are welded, except where flanged connections are required.	9.2.2.2.1.4
		Design features to provide containment	
		Radiation monitors are located downstream of the supply headers and the signal is indicated in the MCR. When the signal exceeds the setpoint, an alarm is transmitted and the CCW surge tank vent valve is closed and interlocked not to open automatically.	9.2.2.5.2

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Objective		System Features	DCD Reference
2	Provide for adequate leak detection capability to provide prompt detection of leakage for any structure, system, or component which has the potential for leakage.	The radiation monitors are installed to detect the leakage of radioactive materials into the CCWS. If leakage from a higher pressure component to the CCWS should occur, the water level of the CCW surge tank increases and an alarm is transmitted to the MCR. If the in-leakage is radioactive, the radiation monitors of the CCWS also indicate in the MCR the increased radiation level and transmit an alarm when the radiation level reaches its set point. After the leak source is identified, the leak is isolated from the CCWS.	1.2.1.5.4.4 9.2.2.3.1
3	Use leak detection methods (e.g., instrumentation, automated samplers) capable of early detection of leaks in areas where it is difficult or impossible to conduct regular inspections (such as for spent fuel pools, tanks that are in contact with the ground, and buried, embedded, or subterranean piping) to avoid release of contamination of the environment.	The leak detection instrumentation is described above.	—
4	Reduce the need to decontaminate equipment and structures by decreasing the probability of any release, reducing any amounts released, and decreasing the spread of the contaminant from the source.	The detection and mitigation features discussed above decrease the probability of releases.	—

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 13 of 52)

Objective		System Features	DCD Reference
5	Facilitate decommissioning by (1) minimizing embedded and buried piping, and (2) designing the facility to facilitate the removal of any equipment and/or components that may require removal and/or replacement during facility operation or decommissioning.	a) Embedded piping is kept to a minimum. b) Process equipment items are accessible for maintenance and replacement. These items are small in volume and may be removed and replaced with minimum cutting into small pieces.	—
6	Minimize the generation and volume of radioactive waste both during operation and during decommissioning (by minimizing the volume of components and structures that become contaminated during plant operation).	All of the features listed above minimize the possibility of contamination of components and structures, and therefore minimize the volume of radioactive waste generated during operation and decommissioning.	—

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Water Systems (Note: The "System Features" column consists of excerpts/summary from the DCD)

Condensate Storage Facility

Objective		System Features	DCD Reference
1	Minimize leaks and spills and provide containment in areas where such events may occur.	The transfer piping running between the CST and the hotwell is single-walled welded stainless steel piping in a coated trench with removable but sealed covers. This design is supplemented by periodic hydrostatic or pressure testing of pipe segments, instrument calibration, and when required, visual inspection and maintenance of piping, trench and instrument integrity, in compliance with the guidance of RG 4.21 and industry operating experience.	9.2.6.2.4
2	Provide for adequate leak detection capability to provide prompt detection of leakage for any structure, system, or component which has the potential for leakage.	Piping in a coated trench with removable but sealed covers, this design is supplemented by periodic hydrostatic or pressure testing of pipe segments, instrument calibration, and when required, visual inspection and maintenance of piping, trench and instrument integrity.	9.2.6.2.4
3	Use leak detection methods (e.g., instrumentation, automated samplers) capable of early detection of leaks in areas where it is difficult or impossible to conduct regular inspections (such as for spent fuel pools, tanks that are in contact with the ground, and buried, embedded, or subterranean piping) to avoid release of contamination of the environment.	Piping in a coated trench with removable but sealed covers, this design is supplemented by periodic hydrostatic or pressure testing of pipe segments, instrument calibration, and when required, visual inspection and maintenance of piping, trench and instrument integrity.	9.2.6.2.4

Water Systems <u>**(Note: The "System Features" column consists of excerpts/summary from the DCD)**</u>

Primary Makeup Water Tanks (PMWTs)

1	Minimize leaks and spills and provide containment in areas where such events may occur.	<u>**Design features to minimize leaks**</u> The piping to and from the PMW Tank is single-walled stainless steel piping designed to run aboveground and penetrates the building wall directly into the tank. This piping is mostly inside the A/B in pipe chases. For piping between buildings, penetration sleeves are provided to collect and direct any leakages back into the building for further processing. The piping may require heat tracing to protect against freezing. The PMWTs employ non-leakage type valves such as diaphragm-type valves, or leak control valves with graphite packing for handling radioactive fluid, or leak-off connection is provided to prevent leakage to environment. Similar piping is provided for the PMW Tanks carrying recycle water back to the A/B. This design is supplemented by operational programs which includes periodic hydrostatic or pressure testing of pipe segments, and visual inspections to maintain piping integrity. A discussion on minimizing radioactive contamination of the system is contained in DCD Section 12.3.1.3. CSF tanks including PMWTs and CST and their associated piping are periodically tested / inspected for leakages.	9.2.6.2.6 9.2.6.4

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 16 of 52)

Process Auxiliaries **(Note: The "System Features" column consists of excerpts/summary from the DCD)**

Process and Post-Accident Sampling Systems

Objective		System Features	DCD Reference
1	Minimize leaks and spills and provide containment in areas where such events may occur.	The Primary Liquid Sampling System (PLSS) is designed to collect liquid samples from the Reactor Coolant System and the auxiliary systems. These samples are transported to a common location in the sampling room of the Auxiliary Building. Limiting the number of locations at which sampling analysis is performed will minimize the spread of contamination within the plant buildings.	9.3.2.2.1
		PLSS sampling is performed via manual operation on an intermittent basis in order to minimize the time during which leaks and spills can occur. The sample lines are purged before each sample is drawn and the purged liquid is returned to the low-pressure end of its own system. This liquid is therefore contained within the sampling system and the contamination of other structures, systems, or components is minimized.	9.3.2.2.1
		Leakage from the Post Accident Sampling System (PASS) outside the containment is collected by the Reactor Building sump tank to prevent the contamination of other systems and areas.	9.3.2.1

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 17 of 52)

Objective		System Features	DCD Reference
2	Provide for adequate leak detection capability to provide prompt detection of leakage for any structure, system, or component which has the potential for leakage.	The Steam Generator Blowdown Sampling System (SGBDSS) is designed to provide detection of primary-to-secondary leakage within the steam generator tubes. The samples taken from the blowdown are monitored for radioactivity as a means to indicate tube leakage. In addition, Condenser vacuum pump exhaust is equipped with radiation monitors to close the steam generator blowdown isolation valves in the event that radiation level exceeds a pre-determined setpoint.	9.3.2.2.5 10.3.3 Figure 10.4.2-1 11.5.2.4.2
3	Use leak detection methods (e.g., instrumentation, automated samplers) capable of early detection of leaks in areas where it is difficult or impossible to conduct regular inspections (such as for spent fuel pools, tanks that are in contact with the ground, and buried, embedded, or subterranean piping) to avoid release of contamination of the environment.	The Primary Gaseous Sampling System (PGSS) is designed to collect representative samples from the containment atmosphere during normal operation. The analysis of these samples will be used to determine the gaseous composition of the containment and will in turn indicate the presence of radioactive material in-leakage.	9.3.2.2.2

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Objective		System Features	DCD Reference
4	Reduce the need to decontaminate equipment and structures by decreasing the probability of any release, reducing any amounts released, and decreasing the spread of the contaminant from the source.	In order to reduce the source volume exposed at the sample panel of the PLSS, the components of this system which contain radioactive liquids are located in shielded compartments including the sample coolers, isolation valves, and system piping.	9.3.2.2.1
		Samples collected by the PLSS are transferred to the sample panel within a ventilated, hooded enclosure in order to confine any leakage or spillage within this system. Any contaminated liquid from leaks or spills will be collected in the sample sink and will be drained to the Waste Holdup Tank to be treated by the Liquid Waste Management System. A similar line is included for the post-accident liquid samples. This will prevent the spread of contamination from the PLSS to other plant systems or equipment.	9.3.2.2.1 9.3.2.2.3
		Samples collected by the PGSS are contained in gaseous sample vessels which are positioned within a filtered vent hood. In addition, residual dew condensation liquid collected on these gaseous sample vessels is collected and routed to the waste holdup tanks to be processed by the Liquid Waste Management System. These design features will assist in decreasing the spread of potentially contaminated process fluids from the PGSS.	9.3.2.2.2
5	Facilitate decommissioning by (1) minimizing embedded and buried piping, and (2) designing the facility to facilitate the removal of any equipment and/or components that may require removal and/or replacement during facility operation or decommissioning.	a) There will be no embedded or buried piping. b) The sample vessels utilized in the PLSS for high-pressure samples are a removable design equipped with quick-disconnect couplings in order to facilitate the removal of the vessels to be brought to the radiochemical laboratory for analysis.	9.3.2.2.1
		Grab sampling points are provided for liquid sample collection as needed. These points include sample vessel connections equipped with quick-disconnect couplings in order to facilitate the removal of the vessels and transport for analysis.	9.3.2.2.6

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Objective		System Features	DCD Reference
6	Minimize the generation and volume of radioactive waste both during operation and during decommissioning (by minimizing the volume of components and structures that become contaminated during plant operation).	The Post-Accident Sampling System (PASS) contains venting which is transferred to the HVAC System. This venting undergoes radiation monitoring. In the event that high radiation levels are detected, the venting stream is re-routed to a line with HEPA and charcoal filters. These HVAC design features will minimize the spread of contamination from the PASS.	9.3.2.2.3
		The SGBDSS is designed to perform the function of detecting high radiation levels in the steam generator blowdown which is an indication of tube leakage. In the event that high radiation is detected, a signal is sent which automatically isolates the steam generator blowdown lines and the steam generator sample lines. This isolation will minimize the spread of contamination from the site of the leakage within the steam generator system.	9.3.2.2.5
		The PLSS, PGSS, PASS and SGBDSS lines, which penetrate the containment, can be isolated through the manipulation of valves either by receipt of a containment isolation signal or by manual actuation. This will prevent the spread of contamination from the containment into any of the sampling sub-systems.	9.3.2.3

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Process Auxiliaries **(Note: The "System Features" column consists of excerpts/summary from the DCD)**

Chemical and Volume Control System (CVCS)

Objective		System Features	DCD Reference
1	Minimize leaks and spills and provide containment in areas where such events may occur.	**Design features to minimize leaks**	
		- The CVCS employs diaphragm-type valves which minimize leaks, where applicable. For components which cannot structurally employ these types of valves, a leak-off connection is provided to prevent leakage to the atmosphere.	9.3.4.2.6.26
		- Heat exchangers are designed with corrosion-resistant materials and radioactive fluid is processed through the tube side.	12.3.1.1.1.2. F
		Design features to provide containment	
		- The CVCS piping that handles radioactive liquid is made of austenitic stainless steel.	9.3.4.2.6.26
		- The piping joints and connections are welded except where flanged connections are required for equipment removal for maintenance and hydrostatic testing.	9.3.4.2.6.29
		- Tank cubicles are coated with non-porous material up to a wall height to contain the entire tank content. The cubicles are equipped with a drainage system to direct any leakage and overflows to sumps with pumps to redirect flow to other tanks.	12.3.1.1.1.2.E

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 21 of 52)

Objective		System Features	DCD Reference
2	Provide for adequate leak detection capability to provide prompt detection of leakage for any structure, system, or component which has the potential for leakage.	- The drainage system is equipped with a liquid detection instrument which can provide early warning for leakage and/or overflow condition to initiate operator actions. - The floors of these cubicles containing radioactive fluid are sloped to facilitate faster drainage and to minimize liquid accumulation, and provided with coating with non-porous material to prevent cross contamination. - On the shell side of heat exchangers, the return header has a radiation monitor to isolate the cooling water system, in the event leakage is detected.	12.3.1.1.1.2.E
3	Use leak detection methods (e.g., instrumentation, automated samplers) capable of early detection of leaks in areas where it is difficult or impossible to conduct regular inspections (such as for spent fuel pools, tanks that are in contact with the ground, and buried, embedded, or subterranean piping) to avoid release of contamination of the environment.	The leak detection system is incorporated in all cubicles in which the tanks contain radioactive fluid (refer to system features for objective #2 above). The tanks include: o Holdup Tanks o Volume Control Tank o Boric Acid Tanks	--

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Objective		System Features	DCD Reference
4	Reduce the need to decontaminate equipment and structures by decreasing the probability of any release, reducing any amounts released, and decreasing the spread of the contaminant from the source.	In addition to the features discussed in items above, the following design specifications and operational procedures are also implemented: o Water chemistry is strictly monitored for primary and secondary systems. In particular, water is treated to control oxygen which is chemically scavenged to minimize potential for corrosion; o Stainless steel will be specified as the materials which are resistant to corrosion. Surface will be polished to facilitate easy decontamination; and o Suitable smooth-surface coatings facilitate the decontamination of potentially contaminated areas and equipment. o Floor drains with properly sloping floors are provided and radioactive and potentially radioactive drainage is separated from non-radioactive drainage. The summation of these design and operational features is designed to reduce the need to decontaminate equipment and structures by decreasing the probability of any release, reducing any amounts released, and decreasing the spread of the contaminant from the source.	9.3.4.1.2.3 9.3.4.2.6 12.3.1.1.2.D 12.3.1.1.2.D
5	Facilitate decommissioning by (1) minimizing embedded and buried piping, and (2) designing the facility to facilitate the removal of any equipment and/or components that may require removal and/or replacement during facility operation or decommissioning.	a) The design uses a drain header to direct potential leakage and spills to the reactor drain tank via a common header. This design minimizes embedded and buried piping within the building foundation slab. No other embedded piping is anticipated in the current design. b) Process equipment items are designed to be accessible for maintenance and replacement. Equipment is designed for extended life and will not need to be replaced. If leak develops, equipment can be accessed for repairs in place. Equipment can be decontaminated to remain in the cubicles until decommissioning, during which time they can be cut into smaller pieces for disposal.	--

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Objective		System Features	DCD Reference
6	Minimize the generation and volume of radioactive waste both during operation and during decommissioning (by minimizing the volume of components and structures that become contaminated during plant operation).	All of the features listed above minimize the possibility of contamination of components and structures, and therefore minimize the volume of radioactive waste generated during operation and decommissioning.	—

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Process Auxiliaries **(Note: The "System Features" column consists of excerpts/summary from the DCD)**

Air Conditioning, Heating, Cooling, and Ventilation Systems

Objective		System Features	DCD Reference
1	Minimize leaks and spills and provide containment in areas where such events may occur.	**Design features to minimize leaks** - Air distribution ductwork is leak-tested in accordance with the Sheet Metal and Air Conditioning Contractors' National Association (SMACNA) technical manual "HVAC Air Duct Leakage Test Manual" and American Society of Mechanical Engineers, ASME N510, AG-1 Section SA and TA. - All non-safety related ductwork located in areas containing safety related components (Reactor Building, the Fuel Handing Area and the Power Source Building) shall be designed to seismic category II. They will remain intact and functional and prevent interaction with safety-related SSC's in these areas. **Design features to provide containment** - The penetration and the safeguard component areas supply and exhaust duct work are isolated in order that operation of the annulus emergency exhaust system can maintain a negative pressure and mitigate the release of airborne fission product to the atmosphere - The auxiliary building HVAC system discharge duct is isolated in order to prevent backflow of discharge air from the annulus emergency exhaust system into the auxiliary building HVAC system.	9.4.1.4 9.4.3.4 9.4.5.4 9.4.3.1.1.1 9.4.3.1.1.1

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 25 of 52)

Objective	System Features	DCD Reference
	- The ventilation systems in potentially contaminated areas maintain airflow from areas of low radioactivity to areas of potentially higher radioactivity.	9.4.3.1.2.1 12.3.3.3
	- The auxiliary building HVAC system controls exhaust fan airflow continuously and automatically at a predetermined value higher than the supply fan airflow to maintain a slightly negative pressure in the controlled areas relative to the outside atmosphere. This minimizes exfiltration from the radiological controlled areas during normal plant operation. The exhaust airflow through the plant's vent stack is a radiologically monitored path.	9.4.3.1.2.1 12.3.3.3
	- Upon receipt of the ECCS actuation signal, the auxiliary building HVAC system discharge duct is automatically isolated by the equipment class 2, seismic category I isolation dampers in order to prevent backflow of discharge air from the annulus emergency exhaust system into the auxiliary building HVAC system.	9.4.3.3.1
	- Ventilation ducts are designed to minimize the buildup of radioactive contamination within the ducts.	
	- Air cleaning systems are utilized.	12.3.3.3
	- The HVAC system design facilitates the replacement of the filter elements.	
	- The tanks vent line is either open to the cubicle or connected directly to the ventilation system.	12.3.3.3 12.3.1.1.1.2.K 12.3.1.1.1.2.E

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 26 of 52)

Objective		System Features	DCD Reference
2	Provide for adequate leak detection capability to provide prompt detection of leakage for any structure, system, or component which has the potential for leakage.	The auxiliary building HVAC system controls exhaust fan airflow continuously and automatically at a predetermined value higher than the supply fan airflow to maintain a slightly negative pressure in the controlled areas relative to the outside atmosphere. This minimizes exfiltration from the radiological controlled areas during normal plant operation. The exhaust airflow through the plant's vent stack is a radiologically monitored path. Airborne radiation monitors in the exhaust ductwork from areas like the fuel handling area, reactor building controlled areas, auxiliary building controlled areas and access control building areas will alarm in the control room. The control room operators will remotely from the control room manually isolate the supply and exhaust ductwork from the areas as needed and redirect airflow to the containment low volume purge exhaust system, filters, which are then vented through the plant vent stack, during normal plant operation. The exhaust airflow through the plant's vent stack is a radiologically monitored path.	9.4.3.1.2.1 12.3.3.3 9.4.3.2.1 12.3.4.2

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 27 of 52)

Objective		System Features	DCD Reference
3	Use leak detection methods (e.g., instrumentation, automated samplers) capable of early detection of leaks in areas where it is difficult or impossible to conduct regular inspections (such as for spent fuel pools, tanks that are in contact with the ground, and buried, embedded, or subterranean piping) to avoid release of contamination of the environment.	The MCR HVAC system equipment and components are provided with proper access for initial and periodic inspections and maintenance during normal operation.	9.4.1.4
		The Auxiliary Building, Non-Class 1 E Electrical Room, Main Steam/Feedwater Piping Area, and TSC HVAC systems are designed to provide accessibility to system components for adjustment, maintenance and periodic inspection and testing of the system components to assure proper equipment function and reliability and system availability.	9.4.3.1.2.1, 9.4.3.1.2.2 9.4.3.1.2.3, 9.4.3.1.2.4
		The ESF ventilation system is designed to provide accessibility to system components for adjustment, maintenance, and periodic inspection and testing of the system components.	9.4.5.1.2 9.4.5.4
		Therefore, there are no areas where it is difficult or impossible to conduct regular inspections, testing, maintenance and adjustments.	

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 28 of 52)

Objective		System Features	DCD Reference
4	Reduce the need to decontaminate equipment and structures by decreasing the probability of any release, reducing any amounts released, and decreasing the spread of the contaminant from the source.	Upon receipt of the ECCS actuation signal, the penetration and the safeguard component areas are automatically isolated by the equipment class 2, seismic category I isolation dampers in order that operation of the annulus emergency exhaust system maintains a negative pressure and mitigates the release of airborne fission products to the atmosphere. This low probability of release decreases need to decontaminate equipment and structures, and ensures contaminants do not spread from the source. The auxiliary building HVAC system controls exhaust fan airflow continuously and automatically at a predetermined value higher than the supply fan airflow to maintain a slightly negative pressure in the controlled areas relative to the outside atmosphere. This minimizes exfiltration from the radiological controlled areas during normal plant operation. The exhaust airflow through the plant's vent stack is a radiologically monitored path. The containment low volume purge system during normal plant operation maintains the pressure inside containment. This air is treated through a HEPA and charcoal absorber filter and vented through the plant stack, which is a radiologically monitored path.	9.4.3.3.1 9.4.5.1.1.1 9.4.5.2.1 12.3.3.3 9.4.3.1.2.1 9.4.3.2.1 12.3.3.3
5	Facilitate decommissioning by (1) minimizing embedded and buried piping, and (2) designing the facility to facilitate the removal of any equipment and/or components that may require removal and/or replacement during facility operation or decommissioning.	a) There is no embedded piping or ductwork in the HVAC systems. b) HVAC equipment and components are accessible for maintenance and replacement. These items are small in volume and may be removed and replaced with minimum cutting into small pieces.	--

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 29 of 52)

Objective		System Features	DCD Reference
6	Minimize the generation and volume of radioactive waste both during operation and during decommissioning (by minimizing the volume of components and structures that become contaminated during plant operation).	The ventilation system produces no radioactive waste. The ventilation system is used to contain and minimize contamination and provide a means to monitor airborne contamination. All of the features listed above minimize the possibility of contamination of components and structures, and therefore minimize the volume of radioactive waste generated during operation and decommissioning. However, since the exhaust ductwork for the auxiliary building ventilation system is the major exhaust path for the plant and takes suction from all areas that are contaminated or potentially contaminated, the ductwork will be internally contaminated by the time the plant is decommissioned. The only method of minimizing the contamination of components is to use less ductwork. This will be a design consideration.	—

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(Note: The "System Features" column consists of excerpts/summary from the DCD)

Liquid Waste Management System (LWMS)

Objective		System Features	DCD Reference
1	Minimize leaks and spills and provide containment in areas where such events may occur.	**Design features to minimize leaks**	
		- SSC are designed in accordance with RG 1.143, Table 1	11.2.1.2
		- SSC are specified as welded construction	11.2.1.4
		- Tanks containing liquid are designed in accordance with ANSI 55.6 with 20% safety factor and a minimum of 10% freeboard allowance	11.2.1.2
		- Piping are in accordance with ANS B31.3, butt-welded to minimize leakage	Table 11.2-1
		Design features to minimize spills	
		- Tank levels are protected with High and High-High alarms and interlocks to prevent over fill and spills	Table 11.2-8
		- Tanks are cross-tied to allow fluid to be directed to other tanks for surge flow	11.2.1.2
		- Tanks are equipped with overflow piping to direct any overflow to drainage system	11.2.1.2
		Design features to provide containment	
		- Tank cubicles are coated with non-porous material up to a wall height to contain the entire tank content. The cubicles are equipped with drainage system to direct any leakage and overflows to sumps with pumps to redirect flow to other tanks.	12.3.1.1.1.2.E

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Objective		System Features	DCD Reference
		– In addition, the primary makeup water tanks and refueling water storage auxiliary tank, which are located in a tank house, also have same design features. The walls and floors of the tank house are coated with non-porous material. The floor is sloped towards the drainage pit or funnel. The drainage system is equipped with a liquid detection.	
2	Provide for adequate leak detection capability to provide prompt detection of leakage for any structure, system, or component which has the potential for leakage.	– The drainage system is equipped with a liquid detection instrument which can provide early warning for leakage and/or overflow condition. – The floors of these cubicles containing radioactive fluid are sloped to facilitate faster drainage and to minimize liquid accumulation, and provided with coating with non-porous material to prevent cross contamination.	12.3.1.1.1.2.E
3	Use leak detection methods (e.g., instrumentation, automated samplers) capable of early detection of leaks in areas where it is difficult or impossible to conduct regular inspections (such as for spent fuel pools, tanks that are in contact with the ground, and buried, embedded, or subterranean piping) to avoid release of contamination of the environment.	The leak detection system is incorporated in all cubicles in which the tanks contain radioactive fluid (refer to system features for objective #2 above). The tanks include: o Waste Holdup Tanks o Waste Monitor Tanks o Spent Resin Storage Tanks	—

Objective		System Features	DCD Reference
4	Reduce the need to decontaminate equipment and structures by decreasing the probability of any release, reducing any amounts released, and decreasing the spread of the contaminant from the source.	In addition to the features discussed in items above, the following design specifications and operational procedures are also implemented: o Stainless steel will be specified as the material which is resistant to corrosion. Surface finish will be polished to facilitate easy decontamination; and o Suitable smooth-surface coatings facilitate the decontamination of potentially contaminated areas and equipment. o Floor drains with properly sloping floors are provided and radioactive and potentially radioactive drainage is separated from non-radioactive drainage. o The LWMS and the Solid Waste Management System (SWMS), which employ flexible interconnecting piping for radioactive fluids, are designed with connectors that are incompatible with the connectors for non-radioactive fluids to prevent accidental cross-contamination. The summation of these design and operational features is designed to reduce the need to decontaminate equipment and structures by decreasing the probability of any release, reducing any amounts released, and decreasing the spread of the contaminant from the source.	11.2.2.2 12.3.1.1.2.D 12.3.1.1.2.D

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Objective		System Features	DCD Reference
5	Facilitate decommissioning by (1) minimizing embedded and buried piping, and (2) designing the facility to facilitate the removal of any equipment and/or components that may require removal and/or replacement during facility operation or decommissioning.	a) The design uses a drain header to direct potential leakage and spills to the floor drain sump via a common header. This design minimizes embedded and buried piping within the building foundation slab. No other embedded piping is anticipated in the current design. b) Process equipment items (such as pumps, filters, ion exchangers in the LWMS) are designed with decontamination capability, and to be accessible for maintenance and replacement. Equipment is designed for extended life and will not need to be replaced. If leak develops, equipment can be accessed for repairs in place. Equipment can be decontaminated to remain in the cubicles until decommissioning, during which time they can be cut into smaller pieces for disposal.	--

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Objective		System Features	DCD Reference
6	Minimize the generation and volume of radioactive waste both during operation and during decommissioning (by minimizing the volume of components and structures that become contaminated during plant operation).	The treatment technologies (filtration and ion exchange) use simple but effective design that is industrial proven. These technologies concentrate contaminants in the filters and ion exchange resin and thus minimize waste generation during operation. Instrumentation is provided in the design to control the operation to ensure that the filters and ion exchange resin are efficiently used in order to minimize waste generation. The filter and ion exchange vessels are designed and specified to be stainless steel for resistant to corrosion, polished surfaces for ease of decontamination, equipped with flush water for decontamination during operation and decommissioning purpose. These equipment features greatly reduces worker doses during maintenance and decommissioning activities. Chemicals to be used for plant operation are regulated and controlled in order to minimize the generation of mixed waste. To the extent possible, non-hazardous chemicals (both listed and characteristic) are to be minimized for use. If it cannot be avoided, these and all other chemicals are tracked to prevent misuse and thus to minimize waste generation. Solid wastes generated during maintenance activities are required to be first decontaminated and then sorted at the point of generation. These procedures will be instituted in the Process Control Manual and will be used to minimize waste generation and avoid mixing different wastes into single waste classification. These wastes will be sent to specialized off-site facilities for more efficient and economic processing and disposal.	11.2.1.4

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Containment <u>**(Note: The "System Features" column consists of excerpts/summary from the DCD)**</u>

Objective		System Features	DCD Reference
1	Minimize leaks and spills and provide containment in areas where such events may occur.	The US-APWR containment vessel consists of a prestressed, post-tensioned concrete structure with a cylindrical wall, hemispherical dome, and flat reinforced concrete foundation slab. The inside surface of the structure is lined with carbon steel. The USAPWR reactor and reactor coolant system (RCS) are completely enclosed in the prestressed concrete containment vessel (PCCV). The PCCV is designed to assure essentially no leakage of radioactive materials to the environment, even if a major failure of the reactor coolant system were to occur.	1.1.2
		The concrete shell inner surface is lined with a minimum 1/4-in. carbon steel plate that is anchored to the concrete shell and dome to provide the required pressure boundary leak tightness. Areas around penetrations, support brackets, inner walls, and heavy components bases have thickened steel liner plates.	3.8.1
		The containment is essentially leak tight to ensure that no significant amount of radioactive material can reach the environment, even in the unlikely event of a RCS failure.	6.02
		The containment is a prestressed, post-tensioned concrete structure with a cylindrical wall, a hemispherical dome, and a flat, reinforced concrete foundation slab. To ensure leak tightness during normal operation and under postulated accident conditions, the US-APWR containment is designed and built to safely accommodate an internal pressure of 68 psig.	
		The US-APWR containment is designed to permit periodic leakage rate testing. The periodic leakage rate testing program is the responsibility of any utility that references the US-APWR design for construction and licensed operation.	

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Objective		System Features	DCD Reference
3	Use leak detection methods (e.g., instrumentation, automated samplers) capable of early detection of leaks in areas where it is difficult or impossible to conduct regular inspections (such as for spent fuel pools, tanks that are in contact with the ground, and buried, embedded, or subterranean piping) to avoid release of contamination of the environment.	Leakage testing of the RWSP liner (cladding) is performed in accordance with ASME Code requirements. Inspection criteria are delineated in ASME Code Article CC-5000. Failed inspection areas are repaired in accordance with ASME Code.	6.2.1.6

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(Note: The "System Features" column consists of excerpts/summary from the DCD)

Reactor Coolant System Boundary and Connected Systems

Objective		System Features	DCD Reference
1	Minimize leaks and spills and provide containment in areas where such events may occur.	Controlled leakage shaft seals employ a well-established seal system that has been used in many operating plants. No.2 seal is designed and tested to maintain full system pressure for enough time to secure the pump. In the case of No. 1 seal failure, the No. 1 seal leak-off line is automatically closed. If the No. 1 seal fails during normal operation, the No. 2 seal minimizes leakage rates. No. 2 seal is able to withstand full system pressure and the No. 3 seal ensures the backup function. This ensures that leakage into atmosphere would not be excessive. The RCPB welds are accessible for inservice inspections (ISI) to assess the structural and leak-tight integrity (see Section 5.2). For the RV, a material surveillance program conforming to applicable codes is provided (see Chapter 5, Section 5.3	5.4.1.4.1 3.1.2.5 Criterion 14
2	Provide for adequate leak detection capability to provide prompt detection of leakage for any structure, system, or component which has the potential for leakage.	A leak monitoring system is used to provide early detection of leakage from the reactor coolant pressure boundary. Instrumentation is provided to detect significant leakage from the RCPB with indication in the MCR (see Section 5.2). The reactor coolant pressure boundary (RCPB) leak monitoring system provides a means of detecting and, to the extent practical, identifying the source of reactor coolant leakage and monitoring leaks from the reactor coolant and associated systems. This system provides information which permits the plant operators to take corrective action if a leak is evaluated as detrimental to the safety of the facility.	1.2.1.5.2.1 3.1.2.5 Criterion 14 5.2.5

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Objective		System Features	DCD Reference
		The leak monitoring system is designed in accordance with the requirements of General Design Criterion 30 and the regulatory guidance as identified below: • General Design Criterion 30, "Quality of Reactor Coolant Pressure Boundary" of Appendix A of 10 CFR Part 50, "General Design Criteria for Nuclear Power Plants," to provide a means of detecting and, to the extent practical, identifying the source of reactor coolant leakage. • Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems" (Ref.5.2-15).	5.2.5.1
		Identified leakage other than intersystem leakage, such as pump seal or valve packing, is directed to the C/V reactor coolant drain tank where it is monitored by tank pressure, temperature, and level indications. An important identified leakage path for reactor coolant into other systems is a flow to the secondary side of the steam generator (SG) through the SG tubes. Identified leakage through the SG (primary-to-secondary leakage) is detected by one or more of the following: • Steam generator blowdown water radiation monitor • High sensitivity main steam line monitor • Condenser vacuum pump exhaust line radiation monitor • Liquid samples taken from SG blowdown sampling line.	5.2.5.3
		Indications of unidentified coolant leakage into the containment are provided by an air particulate radioactivity monitor, an airborne gaseous radioactivity monitor, an air cooler condensate flow rate monitoring system, and a containment sump level and flow monitoring system. The sensitivity and response time of leakage detection equipment for unidentified leakage is such that a leakage rate, or its equivalent, of 0.5 gpm can be detected in less than an hour. The methods employed for detecting leakage to the containment from	5.2.5.4

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Objective		System Features	DCD Reference
		unidentified sources are: • Containment sump level • Containment airborne particulate radioactivity • Containment airborne gaseous radioactivity • Condensate flow rate from air coolers. Additionally, humidity, temperature, and pressure monitoring of the containment atmosphere are used for alarms and indirect indication of leakage to the containment. The following leak detection systems instruments will provide the indications of reactor pressure boundary leakage in the MCR with alarms. The alarms will alert the operating personnel to monitor for leakage. Procedures for converting various indications to a common leakage equivalent will be available to operating personnel. Monitors for items A through D below are provided in gallon per minute leakage equivalent. Leakage conversion procedure is to be developed as Operating and Emergency Operating Procedures described in DCD Section 13.5.2.1 to convert various indications to an identified and unidentified common leakage equivalent and leakage rate of change. A. Containment airborne particulate radioactivity monitor (Containment radiation monitor, RMS-RE-040) - airborne particulate radioactivity B. Containment airborne gaseous radioactivity monitor (Containment radiation monitor, RMS-RE-041) - airborne gaseous radioactivity C. Containment air cooler condensate flow rate monitoring system - standpipe level D. Containment sump level and flow monitoring system – sump level F. Containment temperature, pressure, and humidity will only have readouts in the MCR and alarms to indicate occurrence of leakage within the containment. This method is used only to detect leaks and	5.2.5.6 COL 5.2.5.6

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Objective		System Features	DCD Reference
		is not used to quantity leak rates.	
3	Use leak detection methods (e.g., instrumentation, automated samplers) capable of early detection of leaks in areas where it is difficult or impossible to conduct regular inspections (such as for spent fuel pools, tanks that are in contact with the ground, and buried, embedded, or subterranean piping) to avoid release of contamination of the environment.	E. Gross leakage detection methods - charging flow rate, letdown flow rate, pressurizer level, VCT level and reactor coolant temperatures are available as inputs for detection by RCS inventory balance. Containment sump levels and pump operation are also available. Total makeup water flow is available from the plant computer for liquid inventory.	5.2.5.6
4	Reduce the need to decontaminate equipment and structures by decreasing the probability of any release, reducing any amounts released, and decreasing the spread of the contaminant from the source.	Leakage from the RCP is controlled by three shaft seals arranged in series, such that reactor coolant leakage to the containment is essentially zero. The No. 1 seal reduces the leak-off pressure to that of the volume control tank. The No. 2 and 3 leak-off lines route each seal leakage to the containment vessel reactor coolant drain tank (CVDT).	5.4.1.4.9

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Steam Generator Blowdown System <u>(Note: The "System Features" column consists of excerpts/summary from the DCD)</u>

Objective		System Features	DCD Reference
2	Provide for adequate leak detection capability to provide prompt detection of leakage for any structure, system, or component which has the potential for leakage.	The major elements of the SG program in accordance with NEI 97-06 are outlined below: 5. Primary to secondary leak monitoring, which gives operators information needed to safely respond when tube integrity becomes impaired and significant leakage or tube failure occurs. Radioactive contamination of the SGBDS can occur by a primary to secondary leakage in the steam generator. Under normal operating conditions, there is no significant amount of radioactivity in the steam generator blowdown. The isolation valve(s) in each blowdown line provides controls for reducing releases by isolating the affected steam generator blowdown line following a steam generator tube rupture. An inline radiation monitor on the common line from the steam generator blowdown sample lines, facilitate leak detection. The SG blowdown water radiation monitor in the blowdown sample line continuously monitors SG tube leakage. Upon detection of the significant levels of radioactivity, the blowdown flow is also isolated.	5.4.2.2.2 10.4.8.3 10.4.8.5
3	Use leak detection methods (e.g., instrumentation, automated samplers) capable of early detection of leaks in areas where it is difficult or impossible to conduct regular inspections (such as for spent fuel pools, tanks that are in contact with the ground, and buried, embedded, or subterranean piping) to avoid release of contamination of the environment.	Sample blowdown water for chemistry and detect primary-to-secondary leakage with the SG blowdown water radiation monitor. The blowdown samples are used to check the water chemistry of the blowdown water and to detect leakage or failure of a steam generator tube by radiation monitoring. The SGBDS is automatically isolated from the steam generator by closing the isolation valves in the event of an abnormal condition.	1.2.1.5.3.5
		Ground water conditions and issues relative to the US-APWR are site-specific, including monitoring and safeguards requirements to be implemented to the design and operational requirements in RG 4.21.	COLA

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Engineered Safety System Design <u>**(Note: The "System Features" column consists of excerpts/summary from the DCD)**</u>

Objective		System Features	DCD Reference
1	Minimize leaks and spills and provide containment in areas where such events may occur.	*Prestressed Concrete Containment Vessel (PCCV)* - The PCCV is designed to completely enclose the reactor and RCS and assure that essentially no leakage of radioactive materials to the environment would result even if a major failure of the RCS were to occur.	1.2.1.5.4.1

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Flood Protection from External Sources <u>(Note: The "System Features" column consists of excerpts/summary from the DCD)</u>

Objective		System Features	DCD Reference
1	Minimize leaks and spills and provide containment in areas where such events may occur.	Provisions included in the section are as follows; Below grade, the US-APWR nuclear island and other seismic category I and II structures are primarily protected against exterior flooding and the intrusion of ground water by virtue of their thick reinforced concrete walls and base mats. As recommended by NUREG-0800, SRP 14.3.2 (Reference 3.4-4), the external walls below flood level are equal to or greater than two feet thick to protect against water seepage, and penetrations in the external walls below flood level are provided with flood protection features. Construction joints in the exterior walls and base mats are provided with water stops to prevent seepage of ground water. The COL Applicant is to address any additional measures below grade to protect against exterior flooding and the intrusion of ground water into seismic category I buildings and structures. Below-grade exterior wall penetrations such as for piping and conduits have been minimized to reduce the risk of in-leakage and flooding. Where below-grade piping penetrations are necessary, they are designed to preclude water intrusion. Water-tight doors are used as protective barriers to prevent flood waters from spreading to adjacent divisions in various buildings and elevations. Water-tight doors have position indication for closure verification and are periodically inspected and tested to ensure proper functionality.	3.4.1.2

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Objective		System Features	DCD Reference
4	Reduce the need to decontaminate equipment and structures by decreasing the probability of any release, reducing any amounts released, and decreasing the spread of the contaminant from the source.	Flood protection from the failure of the plant systems such as the outside storage tanks and yard piping is achieved using dikes, levees, retention basins, component location, and/or sited grading and drainage. Dikes, levees, and retention basins are provided to retain leaks and spills due to postulated failures of tanks and vessels, when appropriate. Alternatively, external tanks and piping are located sufficiently far away so that their failure does not jeopardize safety-related equipment. This is accomplished by locating external flood sources so that any spillage or leakage is directed away from safety-related equipment by virtue of the site grading and drains, and by locating these items away from exterior doors that could act as a pathway for flood waters. In addition, buried yard piping is located either in pipe tunnels or sufficiently far away so that cracks or breaks will not result in soil erosion that undermines safety related structures or components. In summary, the US-APWR seismic category I and II structures provide hardened protection as defined in RG 1.59 (Reference 3.4-5) against external flooding through such design features as sloped roofs, thick reinforced concrete with special porosity reducing additives, waterproofing, and special sealing of joints and penetrations.	3.4.1.2

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Flood Protection from Internal Sources **(Note: The "System Features" column consists of excerpts/summary from the DCD)**

Objective		System Features	DCD Reference
4	Reduce the need to decontaminate equipment and structures by decreasing the probability of any release, reducing any amounts released, and decreasing the spread of the contaminant from the source.	Water-tight doors are used as protective barriers to prevent flood waters from spreading to adjacent divisions in various buildings and elevations. Water-tight doors have position indication for closure verification and are periodically inspected and tested to ensure proper functionality.	3.4.1.3

Buried Seismic Category I Piping, Conduits, and Tunnels

(Note: The "System Features" column consists of excerpts/summary from the DCD)

Objective		System Features	DCD Reference
5	Facilitate decommissioning by (1) minimizing embedded and buried piping, and (2) designing the facility to facilitate the removal of any equipment and/or components that may require removal and/or replacement during facility operation or decommissioning.	Buried seismic category I piping, conduits, and tunnels are not present in the US-APWR standard plant design.	3.7.3.7

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Residual Heat Removal System <u>(Note: The "System Features" column consists of excerpts/summary from the DCD)</u>

Objective		System Features	DCD Reference
1	Minimize leaks and spills and provide containment in areas where such events may occur.	The reactor coolant discharged from the CS/RHR pump is circulated through the tube side of the CS/RHR heat exchanger, while cooling is provided by circulating CCW through the shell side. The tubes are welded to the tube sheet to prevent leakage of the reactor coolant.	5.4.7.2.2.2
2	Provide for adequate leak detection capability to provide prompt detection of leakage for any structure, system, or component which has the potential for leakage.	The RHRS is provided with a leakage detection system to minimize the leakage from those portions of the RHR system outside of the containment that contain or may contain radioactive material following an accident.	5.4.7.1

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Containment Isolation Systems **(Note: The "System Features" column consists of excerpts/summary from the DCD)**

Objective		System Features	DCD Reference
2	Provide for adequate leak detection capability to provide prompt detection of leakage for any structure, system, or component which has the potential for leakage.	Systems that are including remote manual valve for containment isolation are followings: • Safety injection system. • Containment spray system • Residual heat removal system • Emergency feedwater system • Main steam system • Seal water injection • Post-accident sampling return line • Fire protection water supply system The condition in which containment isolation is needed in safety injection system, containment spray system and residual heat removal system is when leak occurs in these systems. These systems are located in safeguard component area. Leak detection system is installed in each system. Level instruments are installed in each pump compartment sump. In addition, if leak is occurred, operators can notice by pump suction/discharge pressure and pump flow rate. The piping systems penetrating the containment are provided with leak detection, isolation, and containment capabilities. These piping systems are designed with the capability to test, periodically, the operability of the isolation valves and associated apparatus and determine if valve leakage is within acceptable limits. Regulatory Position: **LEAKAGE CONTROL OUTSIDE CONTAINMENT** Leakage detection and leakage control program outside of containment following an accident shall be discussed. US-APWR Design: A pit (sump) with a leak detector installed in each pump compartment and	5.4.7.1 6.2.4.3 Table 6.3-1

Objective		System Features	DCD Reference
		alarms to MCR to prevent significant leakage of radioactive recirculation water from the high head injection system to the reactor building. The high head injection system is designed to have sufficient redundancy and independence to prevent loss of core cooling function during an accident assuming the isolation of the leaked train after leakage is detected.	

Steam and Power Conversion System (Note: The "System Features" column consists of excerpts/summary from the DCD)

Objective		System Features	DCD Reference
1	Minimize leaks and spills and provide containment in areas where such events may occur.	**Radioactivity Protection** Under normal operating conditions, there are no radioactive contaminants of operational concern present in the steam and power conversion system. However, it is possible for the system to become contaminated through steam generator tube leakage. In this event, radiological monitoring of the main condenser air removal system, the gland seal system, the steam generator blowdown system, and the main steam lines will detect contamination and alarm high radioactivity concentrations. A discussion of the radiological aspects of primary-to-secondary system leakage and limiting conditions for operation is contained in Chapter 11. The steam generator blowdown system described in Subsection 10.4.8 serves to limit the radioactivity level in the secondary cycle, below the operational limits.	10.1.2

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Main Steam Supply (Note: The "System Features" column consists of excerpts/summary from the DCD)

Objective		System Features	DCD Reference
1	Minimize leaks and spills and provide containment in areas where such events may occur.	Radioactive contamination of the MSS can occur by a primary side to secondary side leak in the SG. Under normal operating conditions, there are no significant amount of radioactivity in the MSS. Additionally, the MSIVs provide controls for reducing releases by isolating the affected main steam line following a steam generator tube rupture (SGTR).	10.3.3
2	Provide for adequate leak detection capability to provide prompt detection of leakage for any structure, system, or component which has the potential for leakage.	In-line radiation monitors on each steam line, condenser vacuum pump exhaust line radiation monitor, GSS exhaust fan discharge line radiation monitor and the SG blowdown line radiation monitor facilitate leak detection.	10.3.3

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Circulating Water System <u>**(Note: The "System Features" column consists of excerpts/summary from the DCD)**</u>

Objective		System Features	DCD Reference
1	Minimize leaks and spills and provide containment in areas where such events may occur.	Large CWS leaks due to pipe/expansion joint failures is indicated and alarmed in the control room by a loss of vacuum in the condenser shell. The effects of flooding due to a CWS failure, such as the rupture of an expansion joint, assumes that the flow into the T/B comes from both the upstream and downstream side of the break and also conservatively assumes that one system isolation valve does not fully close. This does not result in detrimental effects on safety-related equipment since there is no safety-related equipment in the T/B and the base slab of the T/B is located at grade elevation. Any leakage from the CWS due to tube leakage into the main condenser is detected by the secondary sampling system (SSS). Also, the TCS is maintained at a higher pressure than the non-ESW system (which draws water from CWS) to prevent leakage of the non-ESW into the TCS.	10.4.5.3.4.1 10.4.5.3.4.2
3	Use leak detection methods (e.g., instrumentation, automated samplers) capable of early detection of leaks in areas where it is difficult or impossible to conduct regular inspections (such as for spent fuel pools, tanks that are in contact with the ground, and buried, embedded, or subterranean piping) to avoid release of contamination of the environment.	The CWS is automatically isolated in the event of gross leakage into the turbine building (T/B) condenser area to prevent flooding of the T/B.	10.4.5

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Condensate and Feedwater System (Note: The "System Features" column consists of excerpts/summary from the DCD)

Objective		System Features	DCD Reference
1	Minimize leaks and spills and provide containment in areas where such events may occur.	The CFS is designed to permit appropriate in-service and functional testing of the system and components to ensure structural integrity and leak-tightness.	10.4.7.1.2

Emergency Feedwater System (Note: The "System Features" column consists of excerpts/summary from the DCD)

Objective		System Features	DCD Reference
1	Minimize leaks and spills and provide containment in areas where such events may occur.	The EFWS is designed to permit appropriate in-service and functional testing of the system and components to ensure their structural integrity and leak-tightness.	10.4.9.3

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Auxiliary Steam Supply System

(Note: The "System Features" column consists of excerpts/summary from the DCD)

Objective		System Features	DCD Reference
1	Minimize leaks and spills and provide containment in areas where such events may occur.	The condensate piping from the ASSS drain tank is a single-walled stainless steel pipe run above ground in pipe chases from the A/B to the T/B, and is then connected to double-walled welded stainless steel piping through the T/B wall penetration to the auxiliary boiler. Since this is not a high traffic area, this segment of pipe is run above ground and is slightly sloped so that any leakage is collected in the outer pipe and drained to the auxiliary boiler building. At the auxiliary boiler building end, a leak detection instrument is provided to monitor leak. A drain pipe is provided to direct any drains to the building sump. The steam piping is jacketed with insulation and heat protection. This design is supplemented by operational programs which includes periodic hydrostatic or pressure testing of pipe segments, instrument calibration, and when required, visual inspection and maintenance of piping, trench and instrument integrity.	10.4.11.2.1
2	Provide for adequate leak detection capability to provide prompt detection of leakage for any structure, system, or component which has the potential for leakage.	The auxiliary steam drain monitors the leakage of the radioactive materials from the boric acid evaporator to the condensed water of the ASSS. Monitoring the leakage from the primary side of the evaporator, the radiation monitor is attached to the downstream of the auxiliary steam drain pump. The high alarm of the monitor isolates the pump discharge line and steam supply line from main steam and trips the pump. **Leakage of radioactive materials from primary side in the B/A evaporator.** If there is leakage of radioactive materials from the primary side in the B/A evaporator, the auxiliary steam drain tank pump discharge isolation valve is closed and the auxiliary steam drain pumps are tripped by the auxiliary steam drain monitor high alarm. The high signal is alarmed to the main control room.	10.4.11.1.2 10.4.11.2.1 10.4.11.2.3

produced during normal reactor operation, including AOOs. The radioactive waste management systems (RWMS) are designed to minimize the potential for an inadvertent release of radioactivity from the facility and to assure that the discharge of radioactive wastes is maintained as low as practicable below the regulatory limits of 10 CFR 20 (Reference 3.1-15), and below the guidelines of 10 CFR 50, Appendix I (Reference 3.1-16), during normal operation. The gaseous and liquid RWMS have adequate capacity and redundancy to meet discharge concentration limits of 10 CFR 20 (Reference 3.1-15) during periods of design-basis fuel leakage. The RWMS, the design criteria, and the amounts of estimated releases of radioactive effluents to the environment are described in Chapter 11. The radiation monitoring of discharge paths of the gaseous and liquid radwaste processing systems and isolation on high radiation is also discussed in Chapter 11. **The US-APWR is designed to minimize the release of radioactive materials in accordance with Regulatory Guide 4.21, "Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning" (Ref. 3.1-18). A discussion of the design objectives and operational programs to address these radiological aspects of the system is contained in DCD Section 12.3.1. System and component design features addressing RG 4.21 (Ref. 3.1-18) are summarized in Table 12.3-8.**

3.1.6.2 Criterion 61 – Fuel Storage and Handling and Radioactivity Control

> The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions.

3.1.6.2.1 Discussion

The spent fuel pit cooling and purification system (SFPCS), fuel handling and radioactive waste systems are designed to cool and purify spent fuel pit (SFP) water, to supply borated water, provide shielding, and assure safety under normal and postulated accident conditions. The SFPCS is a two-train system that consists of a closed circuit that includes: heat exchangers, pumps, demineralizers, and filters. The subsystem is designed to run on Class 1E power during a loss of offsite power.

The SFPCS is designed to maintain the water level of the SFP, to prevent uncovering of the stored fuel from leakage due to failure of the piping, and to assure radiation shielding. Additionally, water may be added from several other sources, if required (Subsection 9.1.3). Adequate shielding is provided as described in Chapter 12. Radiation monitoring is provided as discussed in Chapters 11 and 12.

Normal heating ventilation and air conditioning (HVAC) system for the SFP area and purification and cooling system is provided by the auxiliary building (A/B) HVAC System. This HVAC System is described in Chapter 9.

The SFP cooling subsystem provides cooling to remove residual heat from the fuel stored in the SFP. The SFPCS is designed with redundancy, testability, and inspection capability. SSCs are designed and located so that appropriate periodic inspection and testing may be performed.

3.1-16 Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion, "As Low as is Reasonably Achievable" for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents, Domestic Licensing of Production and Utilization Facilities, Energy. Title 10, Code of Federal Regulations, Part 50, Appendix I, U.S. Nuclear Regulatory Commission, Washington, DC.

3.1-17 Measuring, Evaluating, and Reporting Radioactivity in Solid Waste and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water Cooled Nuclear Power Plants. Regulatory Guide 1.21, Rev. 1, U.S. Nuclear Regulatory Commission, Washington, DC, June 1974.

3.1-18 Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning. RG 4.21, Rev.0, U.S. Nuclear Regulatory Commission, Washington, DC, June 2008.

5.2.5 Reactor Coolant Pressure Boundary (RCPB) Leakage Detection

The reactor coolant pressure boundary (RCPB) leak monitoring system provides a means of detecting and, to the extent practical, identifying the source of reactor coolant leakage and monitoring leaks from the reactor coolant and associated systems. This system provides information which permits the plant operators to take corrective action if a leak is evaluated as detrimental to the safety of the facility.

5.2.5.1 Design Bases

The leak monitoring system is designed in accordance with the requirements of General Design Criterion 30 and the regulatory guidance as identified below:

- General Design Criterion 30, "Quality of Reactor Coolant Pressure Boundary" of Appendix A of 10 CFR Part 50, "General Design Criteria for Nuclear Power Plants," to provide a means of detecting and, to the extent practical, identifying the source of reactor coolant leakage.

- Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems" (Ref.5.2-15).

- Regulatory Guide 1.29, "Seismic Design Classification" (Ref.5.2-7).

- **Regulatory Guide 4.21 "Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning" (Ref.5.2-41) provides a means to minimize contamination and control radioactive leakages. A discussion of the design objectives and operational programs to address these radiological aspects of the system is contained in DCD Section 12.3.1. System and component design features addressing RG 4.21 (Ref.5.2-41)) are summarized in Table 12.3-8.**

5.2.5.2 Classification of Leakage

RCPB leakage is classified as either identified or unidentified leakage in accordance with the guidance of position 1 of regulatory guide 1.45.

Identified leakage includes the following:

- Leakage into closed systems such as pump seals or valve packing leaks that are captured, flow metered, and conducted to a sump or collecting tank.

- Leakage into the containment atmosphere for which the location is identified, without interfering with the unidentified leakage detection system or is identified as leakage from other than the RCPB.

- Leakage into auxiliary systems and secondary systems.

Unidentified leakage is all other leakages.

5.2.5.3 Detection of Identified Leakage

Identified leakage other than intersystem leakage, such as pump seal or valve packing, is directed to the C/V reactor coolant drain tank where it is monitored by tank pressure, temperature, and level indications.

5.2-34 Contents of Applications; Technical Information, NRC Regulations Title 10, Code of Federal Regulations, 50.47.

5.2-35 Operation and Maintenance Code Case Acceptability, ASME OM Code.

5.2-36 US-APWR Sump Strainer Performance, MUAP-08001, Revision 2, December 2008.

5.2-37 Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants, Generic Letter 88-05, March 17. 1998.

5.2-38 Control of Preheat Temperature for Welding of Low-Alloy Steel, Regulatory Guide 1.50, Rev.0, May 1973.

5.2-39 Thermal Aging Embrittlement of Cast Austenitic Stainless Steel Components, License Renewal Issue No. 98-0030, May 19, 2000.

5.2-40 Standard Practice for Steel Casting, Austenitic Alloy, Estimating Ferrite Content Thereof, ASTM A 800/A 800M-01, 2006.

5.2-41 Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning. RG 4.21, Rev.0, U.S. Nuclear Regulatory Commission, Washington, DC, June 2008.

embedment in the pit walls and floors, and the embedment are interconnected and drain through the leakage collection pipe to a collection point which is monitored to determine whether leakage is occurring.

The spent fuel pit leakage collection pipes connected to the C-shape embedment are closed by valves or caps located in the collection points. Any leakage from liner plate welds is detected by opening the valves or caps on patrols conducted weekly. To meet the requirements of 10CFR20.1406, the inside of the spent fuel pit leakage collection pipes are inspected using a device such as a fiberscope approximately every refueling outage. Should materials such as accumulated boric acid residue and minerals be detected, the inside of the pipes are cleaned. The spent fuel pool leakage collection pipes are sized to allow cleaning of blockages as specified in RG 4.21, **"Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning"(Ref. 9.1.7-27). A discussion of the design objectives and operational programs to address these radiological aspects of the system is contained in DCD Section 12.3.1. System and component design features addressing RG 4.21 (Ref. 9.1.7-27) are summarized in Table 12.3-8.**

The refueling canal is connected on one side to the SFP. On its opposite side, the refueling canal connects to the spent fuel cask loading pit and to the fuel inspection pit. A weir and gate provide physical isolation of the refueling canal from each of the three pits. All the gates are located above the top elevation of the fuel seated in the spent fuel racks: they are normally closed and only opened as required.

The SFP is not connected to the equipment drain system (Subsection 9.3.3) to preclude unanticipated drainage.

SFP water level and temperature gauges, and an area radiation monitor in the fuel handling area are provided with alarms to the main control room (MCR) and locally.

Normal auxiliary building (A/B) HVAC system provides ventilation for the fuel handling area to maintain the atmospheric pressure in this area slightly negative with respect to outside the building.

The spent fuel racks are composed of individual vertical cells, and several tiers of grid structures which interconnect each cell to rigidly maintain the cell array configuration. Each rack module is vertically supported by a base plate with 5 legs on the pit floor without anchoring. Additionally, each rack cell is vertically supported by a base plate on the pit floor without anchoring. The grid structures are designed such that a fuel assembly cannot be inserted between the cells.

Moderate density racks containing a neutron absorber material are provided in the SFP. Center-to-center spacing of the rack array is 11.1 inches to maintain the required degree of subcriticality as shown in Figure 9.1.2-2.

Materials used in rack construction are compatible with the SFP environment, and surfaces that come into contact with the fuel assemblies are made of annealed austenitic stainless steel, and are smooth (125AA) in accordance with the requirement of ANSI/ANS-57.2. Structural materials are corrosion resistant and will not contaminate the fuel assemblies or pit environment. Metamic is selected the neutron absorber material. Following program for monitoring the effectiveness of neutron poison by incorporating basic tests assures that the subcriticality requirements of the stored fuel array are maintained.

9.1.7-14 "Occupational Safety and Health Standards," Labor. Title 29 Code of Federal Regulations, Part 1910, U.S. Nuclear Regulatory Commission.

9.1.7-15 "Standards for Protection against Radiation," Energy. Title 10, Code of Federal Regulations, Part 20, U.S. Nuclear Regulatory Commission,

9.1.7-16 "Rules for Construction of Nuclear Facility Components," Boiler and Pressure Vessel Code Section III, American Society of Mechanical Engineers, 2001 Edition through the 2003 Addenda.

9.1.7-17 "Shippers – General Requirements for Shipments and Packagings," Transportation. Title 49, Code of Federal Regulations, Part 173, U.S. Nuclear Regulatory Commission, Washington, DC.

9.1.7-18 "Packaging and Transportation of Radioactive Material," Energy. Title 10, Code of Federal Regulations, Part 71, U.S. Nuclear Regulatory Commission, Washington, DC.

9.1.7-19 Single-Failure-Proof Cranes for Nuclear Power Plants. NUREG-0554, U.S. Nuclear Regulatory Commission, Washington, DC, May 1979.

9.1.7-20 Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder). ASME NOG-1, 2004, American Society of Mechanical Engineers.

9.1.7-21 Control of Heavy Loads at Nuclear Power Plants. NUREG-0612, U.S. Nuclear Regulatory Commission, Washington, DC, July 1980.

9.1.7-22 Overhead and Gantry Cranes (Top Running Bridge, Single or Multiple Girder, Top Running Trolley Hoist). ANSI/ASME B30.2-2005, American Society of Mechanical Engineers.

9.1.7-23 American National Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More for Nuclear Materials. American National Standards Institute, ANSI N14.6-1993, American Nuclear Society, IL.

9.1.7-24 Slings. ANSI/ASME B30.9-2003, American Society of Mechanical Engineers.

9.1.7-25 Specifications for Top Running Bridge and Gantry Type Multiple Girder Electric Overhead Traveling Cranes. CMAA Specification No.70, 2000, Crane Manufacturers Association of America, Inc.

9.1.7-26 Thermal-Hydraulic Analysis for US-APWR Spent Fuel Racks, MUAP-09014P (R0) and MUAP-09014NP (R0), Mitsubishi Heavy Industries, Ltd.,June 2009.

9.1.7-27 **Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning. RG 4.21, Rev.0, U.S. Nuclear Regulatory Commission, Washington, DC, June 2008.**

9.2 Water systems

9.2.1 Essential Service Water System

The essential service water system (ESWS) provides cooling water to remove the heat from the component cooling water (CCW) heat exchangers (HXs) and the essential chiller units. The ESWS transfers the heat from these components to the ultimate heat sink (UHS). The UHS is described in Subsection 9.2.5.

9.2.1.1 Design Bases

The ESWS operates during all modes of plant operation and performs safety-related as well as non-safety related functions. The ESWS is designed to meet the relevant requirements of GDC 2, GDC 4, GDC 5, GDC 44, GDC 45, and GDC 46 (Ref. 9.2.11-1).

9.2.1.1.1 Safety Design Bases

The ESWS is designed to the requirements of the overall US-APWR plant design criteria. Specific safety design bases for the ESWS are as follows:

- The system is capable of transferring heat loads from safety-related SSCs to the UHS during normal operating and accident conditions, including LOCA, pursuant to the requirements of GDC 44.

- The system, in conjunction with the plant UHS, is designed to remove heat from the plant auxiliaries required to mitigate the consequences of a design basis event and for safe shutdown, assuming a single failure and one train unavailable due to maintenance coincident with a loss of offsite power pursuant to the requirements of GDC 44.

- ESWS is designed to equipment Class 3 and seismic category requirements, and as such it is designed to remain functional during and following an SSE per RG 1.29.

- The system is designed considering the protection against adverse environmental, operating, and accident conditions that can occur, such as freezing, thermal overpressurization, and waterhammer per RG 1.206.

- The system is designed in **accordance with Regulatory Guide 4.21, "Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning" (Ref. 9.2.11-9)** to detect and preclude release of radioactive contaminants to the environment. Radioactive contaminants may enter the ESWS from the component cooling water system (CCWS). **A discussion of the design objectives and operational programs to address these radiological aspects of the system is contained in DCD Section 12.3.1. System and component design features addressing RG 4.21 (Ref. 9.2.11-9) are summarized in Table 12.3-8.**

- Measures to prevent long-term corrosion and organic fouling in the ESWS are considered pursuant to the requirements in SRP 9.2.1 and RG 1.206.

- Protection against natural phenomena for the safety-related portions are provided such as protection from wind and tornado effects, as described in Section 3.3; flood protection as described in Section 3.4; internal missile protection as described in

- Detect leakage of radioactive material into the system and control leakage of radioactive material out of the system. **The Component Cooling Water system is subjected to the design objectives of RG 4.21, "Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning" as it contains radioactive liquid. A discussion of the design objectives and operational programs to address these radiological aspects of the system is contained in DCD Section 12.3.1. System and component design features addressing RG 4.21 (Ref. 9.2.11-9) are summarized in Table 12.3-8.**

- Prevent long term corrosion that may degrade system performance.

9.2.2.1.2.1 Normal Operation

The CCWS is designed to transfer heat from the plant components required to support normal power operation with one train (pump and heat exchanger) unavailable due to on line maintenance and a single active component failure. The CCWS is sized such that the component cooling water supply temperature to plant components is not more than 100°F. Normal operating heat loads are reactor coolant pump, charging pump, letdown heat exchanger, instrument air, spent fuel pool cooling heat exchanger, sample heat exchanger, seal water heat exchanger, blowdown sample cooler, B.A. evaporator, waste gas compressor, and so on. The CCWS provides sufficient surge tank capacity below the low level alarm to allow for operators to take action.

9.2.2.1.2.2 Normal Plant Cooldown

The CCWSis designed to remove both decay and sensible heat from the core and the reactor coolant system in addition to some normal operating heat loads during the latter stages of plant cooldown. The component cooling water system is sized to reduce the temperature of the reactor coolant system from 350°F at approximately 4 hours after reactor shutdown to 140°F using 4 trains while maintaining the component cooling water supply below 110°F. Failure of one train of CCW with another train unavailable due to maintenance will not prevent achieving cold shutdown conditions. The CCWS continues to provide cooling water to the residual heat removal system throughout the shutdown after cooldown is complete.

9.2.2.1.2.3 Refueling

During refueling, cooling water flow is provided to spent fuel pool heat exchangers to cool the spent fuel pool. For a full core off-load cooling water is also supplied to a normal residual heat removal heat exchanger as part of spent fuel pool cooling. The CCWS maintains the spent fuel pit water temperature below 120°F. System operation is with both CCWS divisions available.

9.2.2.2 System Description

The system flow diagram is shown in Figure 9.2.2-1.

The CCWS is the closed loop system that functions as an intermediate system between the various components cooled by CCWS and the ESWS, (Subsection 9.2.1). The CCWS transfers heat and prevents direct leakage of the radioactive fluid from the components to the ESWS.

The CCWS consists of two independent subsystems. One subsystem consists of trains A & B, and the other subsystem consists of trains C & D, for a total of four trains. Each train has one

9.2.5.4 Inspection and Testing Requirements

The COL Applicant will provide test and inspection details based on type of UHS to be provided. These details will include inspection and testing requirements necessary to demonstrate that fouling and degradation mechanisms are adequately managed to maintain acceptable UHS performance and integrity.

9.2.5.5 Instrumentation Requirements

The COL Applicant will provide the required alarms, instrumentation and controls details based on the type of UHS to be provided.

9.2.6 Condensate Storage Facilities (Demineralized Water, Condensate Storage, and Primary Makeup Water)

The condensate storage facilities (CSF) system consists primarily of three systems:

- Demineralized water system
- Condensate storage and transfer system
- Primary makeup water system

The demineralized water treatment package is not within the scope of this DCD. The demineralized water treatment package supplies demineralized water to demineralized water storage tank (DWST), which in turn supplies demineralized water to the condensate storage tank (CST), primary makeup water tanks (PMWTs), and other users throughout the plant.

The CSF system is shown schematically in Figures 9.2.6-1, 9.2.6-2, and 9.2.6-3.

The condensate storage and transfer system and primary makeup water system of the condensate storage facility are subjected to the design objectives of RG 4.21, "Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning" as it contains radioactive liquid. A discussion of the design objectives and operational programs to address these radiological aspects of the system is contained in DCD Section 12.3.1. System and component design features addressing RG 4.21 (Ref. 9.2.11-9) are summarized in Table 12.3-8.

9.2.6.1 Design Bases

9.2.6.1.1 Safety-Related Design Basis

The CSF system has no safety-related function and therefore has no nuclear safety design basis.

9.2.6.1.2 Power Generation Design Basis

The demineralized water system is designed to provide:

- Sufficient demineralized water volume for makeup of the CST and to meet the demands and usages of the demineralized water in various other plant systems.
- Sufficient water capacity to provide deaerated water to various users.
- Sufficient water capacity to provide demineralized water to various users.

9.2.11-3 Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants, NRC Regulatory Guide 1.26 Revision 4, March 2007.

9.2.11-4 National Primary Drinking Water Standards, Environmental Protection Agency, Title 40, Code of Federal Regulations, 40CFRPart 141.

9.2.11-5 Occupational Safety and Health Standard, Occupational Safety and Health Administration, Department of Labor, Title 29, Code of Federal Regulations, 29CFRPart 1910.

9.2.11-6 American Nuclear Society Standards Committee Working Group, American National Standard for Decay Heat Power in Light Water Reactors, ANS 5.1, August 1979.

9.2.11-7 Electric Power Research Institute Palo Alto, California, Advanced Light Water Reactor Utility Requirements Document, Rev.8.

9.2.11-8 ANSI B31.1 Power Piping Code.

9.2.11-9 Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning. RG 4.21, Rev.0, U.S. Nuclear Regulatory Commission, Washington, DC, June 2008.

- Manual local grab sample provisions

These systems contain equipment to collect representative samples of the various process fluids in a safe and convenient manner and provide the means to monitor the overall plant condition; and those of various plant systems using the collected and analyzed samples. These systems include sample lines, pressure reduction valves, sample coolers, and automatic analysis equipment. Their design adheres to the as-low-as-reasonably-achievable (ALARA) principle during both normal and post-accident conditions. The PLSS, PGSS, and PASS are located in the A/B and, R/B and the access building. The SGBDSS is located in the R/B and the T/B. The SSS is located in the T/B.

When applicable, sampling frequency and analyses requirements for these systems are listed in the technical specifications. Related discussion of sampling systems and components is provided as follows:

Containment hydrogen monitoring and control system: Chapter 6 / Section 6.2.5

Liquid waste management system: Chapter 11 / Section 11.2

Waste gas analyzer: Chapter 11 / Section 11.3

Process and effluent radiological monitoring and sampling systems:
Chapter 11 / Section 11.5

9.3.2.1 Design Bases

The process and post-accident sampling systems serve no safety functions and therefore have no safety design basis, except for providing for containment isolation, which is described in Chapter 6, Section 6.2.4. The process and post-accident sampling systems are designed in accordance with 10CFR50, Appendix A, General Design Criteria (GDC) 1, 2, 13, 14, 26, 41, 60, 63, 64 (Ref. 9.3.7-8); 10 CFR 20.1101(b) (Ref. 9.3.7-10), NUREG-0737, Item III.D.1.1, SRP Section 9.3.2, 10CFR50.34(f)(2)(viii), ~~and~~ 10CFR50.34(f)(2)(xvii) (Ref. 9.3.7-11) **and Regulatory Guide 4.21 (Ref. 9.3.7-13)**.

The Process and Post-Accident Sampling System is subjected to the design objectives of RG 4.21, "Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning" as it contains radioactive liquid drains from the A/B, Access Building, R/B, and T/B. A discussion of the design objectives and operational programs to address these radiological aspects of the system is contained in DCD Section 12.3.1. System and component design features addressing RG 4.21 (Ref. 9.3.7-13) are summarized in Table 12.3-8.

The containment isolation valves in the PLSS, PGSS, PASS and SGBDSS are selected, tested, and located in accordance with GDC 54, 55, and 56 and 10CFR50, Appendix J, Type C Testing.

The PLSS, PGSS, SSS, SGBDSS and PASS equipment and seismic classification are discussed in Section 3.2.

The US-APWR has its own process and post-accident sampling systems and local grab sample provisions.

Sample lines use 3/8 inch stainless steel tubing and flow restricting orifices to prevent excessive reactor coolant loss.

9.3.2.5 Instrumentation Requirements

The process and post accident sampling systems use local pressure, temperature, and flow indicators to facilitate manual operation and verify sample conditions before samples are drawn.

A radiation detector on the SGBDSS continuously monitors the steam generator blowdown system for primary-to-secondary tube leaks. In the event the SGBDSS detector reaches the radiation set-point as discussed in Section 11.5, the blowdown flow path is automatically closed.

The SSS is equipped with continuous analyzers to monitor specific water quality conditions. Certain measurements are used to automatically control pH and corrosion by chemical addition via the main control panel. Indicators and manual controls are provided on the sampling panel to maintain the proper sample conditions of the water entering the analyzers. Grab sample points are also provided for laboratory analysis and to verify analyzer calibration.

9.3.3 Equipment and Floor Drainage Systems

The equipment and floor drainage systems collect liquid waste from equipment and floor drains during all modes of operation and separate the contaminated effluents and transfer them to the processing and disposal systems. Equipment and floor drainage is classified and segregated by the type of waste generated. Liquid waste classification includes:

- Radioactive liquid waste
- Non-radioactive liquid waste
- Chemical and detergent liquid waste
- Oily liquid waste

9.3.3.1 Design Bases

The Equipment and floor drainage systems are designed in accordance with 10CFR50, Appendix A, General Design Criteria (GDC) 2, 4 and 60 (Ref. 9.3.7-8) **and Regulatory Guide 4.21 (Ref. 9.3.7-13)**.

The Equipment and floor drainage system is subjected to the design objectives of RG 4.21, "Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning" as it contains radioactive liquid drains from the A/B, R/B, T/B, C/V and Access Building. A discussion of the design objectives and operational programs to address these radiological aspects of the system is contained in DCD Section 12.3.1. System and component design features addressing RG 4.21 (Ref. 9.3.7-13) are summarized in Table 12.3-8.

9.3.3.1.1 Safety Design Bases

- The equipment and flood drainage systems function to prevent flooding and water accumulation for volume being drained.
- The equipment and floor drainage systems are not safety-related and serve no safety-related function except the isolation valves installed in the drainage piping from engineered safety feature (ESF) equipment rooms.

decontamination waste, regeneration waste, and detergent waste collection system piping is stainless steel. Non-radioactive collection piping is made of stainless steel.

2. Collection sumps: The centrally located sumps collect normal and potentially radioactive liquid waste. The non-radioactive collection sumps are constructed of concrete with a corrosion resistant coating or liner. These sumps are fitted with a vent connected to the ventilation system to remove any potential radioactive gases. The sumps also collect discharge by gravity from areas that are maintained under a slight negative pressure boundary.

   Radioactive sumps are coated with an impermeable epoxy liner (coating) to facilitate drainage and decontamination. The sumps are equipped with stainless steel sump tanks with welded inlet piping. The gaps between the sump and the sump tanks are covered and/or sealed to prevent infiltration of the liquids.

3. Reactor building equipment and floor drains: The R/B equipment and floor drainage piping are arranged so that any ESF equipment room leakage does not penetrate into other ESF equipment rooms. Discharge from each ESF equipment room is drained by gravity to the either of R/B sump tanks. The drainage piping from each ESF room is equipped with a normally closed, manually operated valve, which is located outside the equipment room.

4. Miscellaneous equipment drains: Equipment which may be pressurized during drainage, and which drains via direct or indirect drain connection to the floor drain system, is designed so that the equipment drain discharge flow will not exceed the gravity flow capacity of the drainage header at atmospheric pressure.

5. Floor drains: All floor drains are installed with rims which are flush with the low-point elevation of the finished floor. All floor drains discharge directly into the respective building sumps or sump tanks.

6. Turbine building equipment and floor drains. The non-radioactive liquid wastes generated in the T/B, including equipment and floor drains and leakages are generally collected in the non-radioactive drain sump in the T/B.

   Turbine building sump pumps discharge to the WWS prior to discharge to the environment. When radioactive contamination in the discharge from the sump is detected and alarmed in the MCR, the transfer valve to the WWS is closed. Following operator initiation, the discharge from the sump is sent to the A/B floor drain sump from which it is transferred to the LWMS for processing prior to discharge to the environment.

7. Equipment and floor drains in the containment, except the reactor coolant drain, are collected in the C/V sump via the drain piping. PS/B equipment and floor drains are collected in the R/B non-radioactive sump.

9.3.3.2.3 System Operation

The equipment and floor drainage systems operate during all modes of operation. The various building drains directly to the corresponding collection point by gravity. The sump pump operation is automatic with level switches. These pumps are automatically started or stopped by a level preset by the local instrumentation in the sump or a sump tank. The T/B sump pumps are not required to operate during design base accident.

containment incorporate valves and piping arrangements that meet the containment isolation criteria described in subsection 6.2.4. Containment isolation valves in the CVCS are required to operate under accident conditions to provide containment isolation,as required.

Since the CVCS supplies non-borated water to the RCS, the potential for inadvertent boron dilution events exists. The design feature for preventing an inadvertent boron dilution is described in Subsection 9.3.4.2.7.6.

The charging line is isolated on a safety injection signal and a Pressurizer high water level signal, to terminate unnecessary RCS makeup that can cause an overfilling of the pressurizer and steam generator overfilling during a steam generator tube rupture.

During a SBO, the reactor coolant pumps seal integrity is maintained until the charging pumps are powered from an alternate power source and seal water injection restarts using the normal seal injection flow path.

The CVCS is designed to provide makeup for minor leaks in the RCS. The makeup capability is limited to the leakage equivalent to a pipe break with 3/8 inch inside diameter.

The CVCS does not provide an ECCS function. Therefore, the provision for a leakage detection and control program in accordance with 10 CFR 50.34 (f) (xxvi) does not apply.

CVCS components and piping are compatible with the radioactive fluids they contain and the functions they perform. The equipment classification for the CVCS is contained in Section 3.2.

The CVCS is designed to ensure that the boric acid solution remains soluble. Heat tracing or a heated area with temperature alarms are provided for portions of the system which normally contain 4 wt. % of boric acid solution, to assure that boric acid solution temperature does not go below 65 °F.

The VCT is designed to withstand vacuum conditions to prevent wall inward buckling and failure. The boric acid tanks are provided with vacuum breakers to prevent a vacuum condition. The holdup tanks are provided with sufficient nitrogen gas supply to prevent vacuum condition.

The CVCS is designed in accordance with the requirements of 10 CFR 50, Appendix A, GDCs are GDC 1, 2, 14, 33, 60, and 61. **The CVCS is subjected to the design objectives of RG 4.21, "Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning" as it contains radioactive fluids. A discussion of the design objectives and operational programs to address these radiological aspects of the system is contained in DCD Section 12.3.1. System and component design features addressing RG 4.21 (Ref. 9.3.7-13) are summarized in Table 12.3-8.**

The protection of safety-related portions of CVCS against natural phenomena and internal missiles is addressed in the following sections in Chapter 3:

Section 3.3, Wind and tornado loadings;

Section 3.4, Water level (Flood) protection;

Section 3.5, Missile protection;

COL 9.3(7) Deleted

9.3.7 References

9.3.7-1 American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III

9.3.7-2 Loss of All Alternating Current Power, NRC Regulations Title 10, Code of Federal Regulations, CFR Part 50.63.

9.3.7-3 Specifications for Air. ANSI/CGA G-7.1, American National Standards Institute.

9.3.7-4 U.S. Nuclear Regulatory Commission, Evaluation of Air-Operated Valves at U.S. Light-Water Reactors, NURGE-1275 Vol.13, February 2000

9.3.7-5 Quality Standards for Instrument Air. ANSI/ISA-S7.3-R 1981, American National Standards Institute/Instrument Society of America, 1981.

9.3.7-6 Seismic Design Classification. Regulatory Guide 1.29, Rev. 4, U.S. Nuclear Regulatory Commission, March 2007.

9.3.7-7 Preoperational Testing of Instrument and Control Air Systems. Regulatory Guide 1.68.3, U.S. Nuclear Regulatory Commission, April 1982.

9.3.7-8 General Design Criteria for Nuclear Power Plants, NRC Regulations Title 10, Code of Federal Regulations, CFR Part 50, Appendix A.

9.3.7-9 Pressure Vessel, ASME Boiler and Pressure Vessel Code Division 1, Section VIII

9.3.7-10 Radiation protection programs NRC Regulations Title 10, Code of Federal Regulations, CFR Part 20.

9.3.7-11 Contents of Applications; Technical Information, NRC regulations Title 10, Code of Federal Regulations, CFR Part 50.34.

9.3.7-12 Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs, SECY-93-087, U.S. Nuclear Regulatory Commission, letter issued April 2, 1993 and staff requirements memoranda issued July 21, 1993.

9.3.7-13 Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning. RG 4.21, Rev.0, U.S. Nuclear Regulatory Commission, Washington, DC, June 2008.

9.4 Air Conditioning, Heating, Cooling, and Ventilation Systems

This section describes the heating, ventilation and air conditioning (HVAC) systems serving the plant during normal and emergency conditions including SBO. HVAC systems are designed to provide suitable environment for plant equipment and personnel. Ventilation zones, air distribution and airflows migration are configured and arranged so that the ventilation air is drawn from the clean areas to areas of potentially greater radioactive contamination to a final filtration and exhaust systems discharging to the plant vent stack.

The HVAC systems airflow diagrams are shown on Figures 9.4.1-1 through 9.4.6-1. The area temperature and relative humidity during the plant normal and emergency condition are described in Table 9.4-1.

The following are the reference sections where the various HVAC and related systems are covered:

Title	Section
Chilled Water System	9.2.7
Main Control Room HVAC System	9.4.1
Spent Fuel Pool Area Ventilation System	9.4.2
Auxiliary Building Ventilation System	9.4.3
Turbine Building Area Ventilation System	9.4.4
Engineered Safety Feature Ventilation System	9.4.5
Containment Ventilation System	9.4.6

The Main Control Room Heating, Ventilation and Air Conditioning System is subjected to the design objectives of RG 4.21, "Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning" as it contains airborne radioactive material. A discussion of the design objectives and operational programs to address these radiological aspects of the system is contained in DCD Section 12.3.1. System and component design features addressing RG 4.21 (Ref. 9.4.8-27) are summarized in Table 12.3-8. RG 4.21 is also applicable to the Auxiliary Building Ventilation System and the Engineered Safety Feature Ventilation System.

9.4.1 Main Control Room Heating, Ventilation and Air Conditioning System

The MCR HVAC System is designed to provide and control the proper environment in the MCR and other areas within the control room envelope (CRE) as defined in Chapter 6, Section 6.4. The MCR HVAC system complies with:

- 10 CFR 50, Appendix A, GDC 2, 3, 4, 19
- 10 CFR 50.63
- RGs, 1.29, 1.52, 1.78, 1.155, 1.196, 1.197, **and 4.21**
- ANSI/ANS 51.1, 59.2
- ASME N509, N510, AG-1
- IEEE 323, 344, 603

9.4.8-11 "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations," IEEE Std 603-1998.

9.4.8-12 "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations," IEEE Std 323™-2003.

9.4.8-13 "IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations," IEEE Std 344™-1987.

9.4.8-14 American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section VIII.

9.4.8-15 "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants," Regulatory Guide (RG) 1.140-2001, Revision2.

9.4.8-16 "Laboratory Methods of Testing Fans for Rating," ANSI/AMCA 210-2007.

9.4.8-17 "Laboratory Methods of Testing Air Circulator Fans for Rating," ANSI/AMCA 230-1999.

9.4.8-18 "Industrial Process / Power Generation Fans: Establishing Performance Using Laboratory Models," ANSI/AMCA 802-2002

9.4.8-19 "Gravimetric and Dust Spot procedures for Testing Cleaning Devices Used in General Ventilation for Removing Particulate Matter," ASHRAE 52.1-1992.

9.4.8-20 "Method of Testing General Ventilation Air-Cleaning Devices for Removal Efficiency by Particle Size," ASHRAE 52.2-2007.

9.4.8-21 "Forced-Circulation Air-Cooling and Air-Heating Coils," ARI 410-2001.

9.4.8-22 "Performance Rating of Room Fan-coils," ARI 440-2005.

9.4.8-23 "1999 Standard for Central Station Air-Handling Units," ANSI/ARI 430-1999

9.4.8-24 "HVAC Air Duct Leakage Test Manual – First Edition; Technical Research Update – 92," SMACNA 1143-1985.

9.4.8-25 "HVAC Systems Testing, Adjusting and Balancing – Third Edition," SMACNA 1780 – 2002

9.4.8-26 International Mechanical Code, 2003 Edition.

9.4.8-27 Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning. RG 4.21, Rev.0, U.S. Nuclear Regulatory Commission, Washington, DC, June 2008.

10.4.8 Steam Generator Blowdown System

The steam generator blowdown system (SGBDS) assists in maintaining secondary side water chemistry within acceptable limits during normal plant operation and during anticipated operational occurrences (AOO) due to the main condenser in leakage or primary to secondary steam generator tube leakage. This is done by removing impurities concentrated in steam generators by continuous blowdown of secondary side water from the steam generators. The system processes blowdown water from all steam generators, as required.

The SGBDS is subjected to the design objectives of RG 4.21, "Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning" as it may contains radioactivity due to the main condenser in leakage and/or primary to secondary steam generator tube leakage. A discussion of the design objectives and operational programs to address these radiological aspects of the system is contained in DCD Section 12.3.1. System and component design features addressing RG 4.21 (Ref. 10.4-20) are summarized in Table 12.3-8.

10.4.8.1 Design Bases

10.4.8.1.1 Safety Design Bases

The safety-related design bases of the SGBDS are as follows:

- The system is provided with a containment isolation valve in each blowdown line from the steam generators.
- The system is provided with two isolation valves in series. These valves isolate the secondary side of the steam generator to preserve the steam generator inventory. This provides a heat sink for a safe shutdown or to mitigate consequences of a design-basis accident.
- The SGBDS performs its safety-related function assuming a single active component failure coincident with the loss-of-offsite or onsite power.
- Piping and valve up to and including the outside containment isolation valve, are designed to ASME Code, Section III (Reference 10.4-8), Class 2, and Seismic Category I requirements. The blowdown system piping and valve from the outlet of the containment isolation valve up to and including first restraint located in the main steam/feedwater piping area are designed in accordance with ASME Code, Section III (Reference 10.4-8), Class 3 and Seismic Category I requirements.
- The safety-related portion of the SGBDS is designed to withstand the effects of a safe-shutdown earthquake and to perform its intended function following a DBA. The system is protected against wind and tornado effects as described in Section 3.3, flood protection as described in Section 3.4, and missile protection as described in Section 3.5, seismic design as described in Section 3.7 and fire protection as described in Subsection 9.5.1.
- The SGBDS safety-related portions constructed in accordance with ASME Section III (Reference 10.4-8), Class 2 and Class 3 requirements are provided with access to welds and removable insulation from areas required for in service inspection in accordance with ASME Section XI (Reference 10.4-12).

10.4-13 U.S. Nuclear Regualtory Commission, Clarification of TMI Action Plan Requirements, NUREG-0737.

10.4-14 U.S. Nuclear Regualtory Commission, Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant. Accidents in Westinghouse – Designed Operating Plants, NUREG-0611, January 1980.

10.4-15 Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants, 10CFR Part 50.62.

10.4-16 Manual Initiation of Protective Actions, Regulatory Guide 1.62 Rev.0, October 1973.

10.4-17 U.S. Nuclear Regualtory Commission, DESIGN REQUIREMENTS OF THE RESIDUAL HEAT REMOVAL SYSTEM, NUREG-0800 Branch Technical Position RSB 5-1.

10.4-18 Design Guidelines for Auxiliary Feedwater System Pump Drive and Power Supply Diversity for Pressurized Water Reactor Plants, NUREG-0800 Branch Technical Position BTP 10-1.

10.4-19 Electric Power Research Institute, PaloAlto, Advanced Light Water Reactor Utility Requirements Document, Rev.8 California.

10.4-20 Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning. RG 4.21, Rev.0, U.S. Nuclear Regulatory Commission, Washington, DC, June 2008.

- The LWMS is designed to meet the requirements of 10 CFR 50, Appendix A (Ref. 11.2-4) Criteria 60, 61, and 64 and the guidance of RG 1.143, (Ref. 11.2-3) so that waste can be successfully processed even during natural phenomena events and external man-induced hazard events.

- The LWMS is designed to process liquid waste generated from normal operation. Radwaste systems normally utilize treated effluent for operations such as sluicing and line flushing to minimize effluent discharge. In the event that there is excess water, or that the treated effluent does not meet recycled water quality specifications, the water is discharged after sampling and analysis confirms the concentration limits of 10 CFR 20 (Ref. 11.2-1). The release is controlled in accordance with 10 CFR 50.34a (Ref. 11.2-5).

- **The LWMS is subjected to the design objectives of RG 4.21, "Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning" as it contains radioactive liquid from the plant. A discussion of the design objectives and operational programs to address these radiological aspects of the system is contained in DCD Section 12.3.1. System and component design features addressing RG 4.21 (Ref. 11.2-24) are summarized in Table 12.3-8.**

- The quality assurance program (QAP) is designed so that the equipment and the installation of the equipment are in accordance with the codes and standards in Table 1 of RG 1.143 (Ref. 11.2-3). The QAP is designed in accordance with ANSI/ANS 55.6 (Ref. 11.2-6).

- The LWMS is designed to operate continuously during normal operating condition and AOOs. For equipment sizing and process capability determination, the LWMS is designed to process the maximum design basis input in one week, assuming 40 hours work week, or processing one tank of liquid waste in one operating shift, whichever is controlling. When excessive wastes are accumulated during normal operation, an additional processing operation can be planned by plant personnel to support overall plant operation.

- The plant is designed in accordance with applicable codes. The QAP assures that the plant is built, maintained, and operated in accordance with the government codes and regulations. This demonstrates that 10 CFR 20.1406 (Ref. 11.2-7) is being implemented.

11.2.1.3 Other Design Considerations

In addition to the listed design criteria, the following considerations are satisfied:

- The LWMS performs no function related to the safe shutdown of the plant. The system's failure does not adversely affect any safety-related system or component. Therefore, the LWMS is not safety-related and performs no safety functions.

- The reactor coolant drainage system is inside the containment and performs no operations relating to the safe shutdown of the plant. However, the containment isolation valves associated with the discharge line from the tank perform a safety function which is discussed in Chapter 6, Section 6.2.

- Pre-operational tests for the LWMS are discussed in Chapter 14, Section 14.2. Thereafter, subsystems and individual components are tested as needed.

11.2-15 Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I, Regulatory Guide 1.109, Rev. 1, U.S. Nuclear Regulatory Commission, Washington, DC, October 1977.

11.2-16 Preparation of Radiological Effluent Technical Specification for Nuclear Power Plants. NUREG-0133. U.S. Nuclear Regulatory Commission, Washington, DC., October 1978.

11.2-17 Postulated Radioactive Releases due to Liquid-Containing Tank Failures. Branch Technical Position 11-6, NUREG-0800, U.S. Nuclear Regulatory Commission, Washington DC., March 2007.

11.2-18 Estimating Aquatic Dispersion of Effluent from Accidental and Routine Releases for the Purpose of Implementing Appendix I, Regulatory Guide 1.113, Rev. 1,U.S. Nuclear Regulatory Commission, Washington DC., April 1977.

11.2-19 Compliance with Dose Limits for Individual Members of the Public Title 10, Code of Federal Regulations, 10 CFR Part 20.1302.

11.2-20 Environmental Radiation Protection Standards for Nuclear Power Operations Title 40, Code of Federal Regulations, 40 CFR Part 190.

11.2-21 Cost-Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Reactors. Regulatory Guide 1.110, March 1976.

11.2-22 Standard Guide for Establishing Procedures to Qualify and Certify Inspection Personnel for Coating Work in Nuclear Facilities, American Society for Testing and Materials, ASTM D 4537-04a.

11.2-23 Standard Guide for Condition Assessment of Coating Service Level Coating Systems in Nuclear Power Plants, American Society for Testing and Materials, ASTM D 5163-08.

11.2-24 Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning. RG 4.21, Rev.0, U.S. Nuclear Regulatory Commission, Washington, DC, June 2008.

Table 1.9.1-2 US-APWR Conformance with Division 4 Regulatory Guides (sheet 2 of 2)

Reg Guide Number	Title	Status	Corresponding Chapter/Section/ Subsection
4.15	Quality Assurance for Radiological Monitoring Programs (Inception through Normal Operations to License Termination) -- Effluent Streams and the Environment (Rev. I-2, March 2007)	Not applicable. RG applies to a site-specific operational program that will be the responsibility of the COL Applicant.	N/A
4.16	Monitoring and Reporting Radioactivity in Releases of Radioactive Materials in Liquid and Gaseous Effluents from Nuclear Fuel Processing and Fabrication Plants and Uranium Hexafluoride Production Plants (Rev. 1, December 1985)	Not applicable. RG applies to non-reactor production facilities.	N/A
4.17	Standard Format and Content of Site Characterization Plans for High-Level-Waste Geologic Repositories (Rev. 1, March 1987)	Not applicable. RG applies to waste repositories.	N/A
4.18	Standard Format and Content of Environmental Reports for Near-Surface Disposal of Radioactive Waste (Rev. 0, June 1983)	Not applicable. RG applies to waste disposal sites.	N/A
4.19	Guidance for Selecting Sites for Near-Surface Disposal of Low-Level Radioactive Waste (Rev. 0, August 1988)	Not applicable. RG applies to waste disposal sites.	N/A
4.20	Constraint on Releases of Airborne Radioactive Materials to the Environment for Licensees other than Power Reactors (Rev. 0, December 1996)	Not applicable. RG applies to non-reactor facilities.	N/A
4.21	Minimization of Contamination and radioactive Waste Generation: Life Cycle Planning (Rev. 0, June 2008)	Conformance with exceptions. Programmatic / operational aspects is not applicable to US-APWR design certification, but implemented by the COL Applicant.	Chapters 1, 3, 4, 5, 6, 9, 10, 11 and 12. Table 12.3-8 provides cross-reference where RG 4.21 features are discussed in the DCD

12.03-12.04-94

The DWST, CST, and the PMWTs are non-safety related and non-seismic (Section 3.2.). These tanks have no safety-related function and failure of their structural integrity would not impact the seismic category I SSCs or cause adverse system interaction. A dike is provided for the PMWTs and CST for mitigating the environmental effects of system leakage or storage tank failure.

The CSF system is shown schematically in Figures 9.2.6-1, 9.2.6-2 and 9.2.6-3.

9.2.6.2.1 Demineralized Water Storage Tank

The DWST is the normal source of demineralized water for supplying water CST, the secondary side chemical injection system, condensate polishing system and the emergency feedwater pits. It is also the normal source for supplying deaerated water to primary makeup water tanks and various primary system users, as shown in Figure 9.2.6-1. The DWST also supplies demineralized water to other users, as shown in Figure 9.2.6-2. Makeup to the CST is provided from the DWST.

Design parameters of the DWST are shown in Table 9.2.6-1.

9.2.6.2.2 Demineralized Water Transfer Pumps

Two 100% capacity demineralized water transfer pumps are provided. The demineralized water transfer pumps take suction from the DWST and discharge into a header that supplies demineralized water to various plant users, as shown in Figure 9.2.6-1. Design parameters of the demineralized water transfer pumps are shown in Table 9.2.6-1.

9.2.6.2.3 Deaeration Package

The deaeration package reduces the oxygen concentration of the demineralized water.

9.2.6.2.4 Condensate Storage Tank

The CST is the normal source of water for make up to certain plant systems including the main condenser. The CST is a source of water for supply to various locations such as areas near equipment that need water for maintenance and drain tanks. Makeup to the CST is provided from the DWST. The CST overflow goes to a dike which is provided to control the release of chemicals and radioactive materials.

The transfer piping running between the CST and the hotwell is single-walled welded stainless steel piping in a coated trench with removable but sealed covers. This design is supplemented by periodic hydrostatic or pressure testing of pipe segments, instrument calibration, and when required, visual inspection and maintenance of piping, trench and instrument integrity, in compliance with the guidance of RG 4.21 and industry operating experience. Design and system features addressing RG 4.21 are captured in Section 12.3.1.3 of the DCD.

Design parameters of the CST are shown in Table 9.2.6-1.

The water chemistry in the CST is maintained in accordance with Table 9.2.6-2.

9.2.6.2.5 Condensate Transfer Pumps

Two 100% capacity condensate transfer pumps are provided. The condensate transfer pumps take suction from the CST and supply condensate to the condenser hotwell and various other users throughout the plant as shown in Figure 9.2.6-1. Design parameters of the condensate transfer pumps are shown in Table 9.2.6-1.

9.2.6.2.6 Primary Makeup Water Tanks

Two 140,000 gallon capacity PMWTs are provided. Each tank is provided with a diaphragm that is in continuous contact with the tank water to prevent absorption of oxygen from air. The top of the diaphragm is blanketed with deaerated, demineralized water. The tanks receive deaerated, demineralized water from the DWST. They also receive distilled water discharged from the boric acid evaporator (subsection 9.3.4). Normally, one tank supplies water to the users, while the other tank is standby. Each tank has sufficient capacity to serve all users. Each tank is provided with level and other instrumentation as shown in Figure 9.2.6-2. Design parameters of the PMWT are shown in Table 9.2.6-1.

The piping to and from the PMW Tank is single-walled stainless steel piping designed to run aboveground and penetrates the building wall directly into the tank. This piping is mostly inside the A/B in pipe chases. For piping between buildings, penetration sleeves are provided to collect and direct any leakages back into the building for further processing. The piping may require heat tracing to protect against freezing. The PMWTs employ non-leakage type valves such as diaphragm-type valves, or leak control valves with graphite packing for handling radioactive fluid, or leak-off connection is provided to prevent leakage to environment. Similar piping is provided for the PMW Tanks carrying recycle water back to the A/B. This design is supplemented by operational programs which includes periodic hydrostatic or pressure testing of pipe segments, and visual inspections to maintain piping integrity. A discussion on minimizing radioactive contamination of the system is contained in DCD Section 12.3.1.3.

9.2.6.2.7 Primary Makeup Water Pumps

Two 100% capacity primary makeup water pumps are provided. The pumps take suction from the PMWT and supply deaerated, demineralized water to plant users as shown in Figure 9.2.6-2. Each pump is a centrifugal pump with 275 gpm capacity. Design parameters of the primary makeup water pumps are shown in Table 9.2.6-1.

9.2.6.3 Safety Evaluation

The CSF system has no safety-related function, and therefore requires no nuclear safety evaluation.

9.2.6.4 Inspection and Testing and Inspection Requirements

The initial preoperational acceptance testing demonstrates proper equipment functioning and system operation. The system's normal functionality is demonstrated by the continuous use during normal plant operation in accordance with the requirement of chapter 14. **CSF tanks including PMWTs and CST and their associated piping are periodically tested / inspected for leakages.**

9.2.6.5 Instrumentation Requirements

The condensate storage facilities are provided with instrumentation, as shown in Figures 9.2.6-1, 9.2.6-2 and 9.2.6-3 to monitor, control, and perform manual or automatic system functions and protect system components.

The CSF System contains a number of automatic on/off valves for its operation in an automatic, semi-automatic, or manual mode.

9.2.6.5.1 Pressure Indicators

Local pressure indication is provided for the pumps.

9.2.6.5.2 Level Transmitters and Level Switches

Level transmitter and associated signal processor units are provided to monitor and indicate water level in the storage tanks. The level in each storage tank is measured and indicated locally and in the MCR. High and low levels are alarmed in the MCR.

plant normal operation, or to the auxiliary boiler during the period in which the main steam is not available.

- Boric acid (B/A) evaporator
- B/A batching tank
- Non safety-related HVAC equipment

The ASSS supplies steam for plant system heating when main steam is not available. The auxiliary boiler takes condensate makeup from the auxiliary steam drain tank inside the A/B, or from the condensate storage tank (CST) in the yard. The auxiliary boiler is located in the yard near the plant area. The condensate piping from the ASSS drain tank is a single-walled stainless steel pipe run above ground in pipe chases from the A/B to the T/B, and is then connected to double-walled welded stainless steel piping through the T/B wall penetration to the auxiliary boiler. Since this is not a high traffic area, this segment of pipe is run above ground and is slightly sloped so that any leakage is collected in the outer pipe and drained to the auxiliary boiler building. At the auxiliary boiler building end, a leak detection instrument is provided to monitor leak. A drain pipe is provided to direct any drains to the building sump. The steam piping is jacketed with insulation and heat protection. This design is supplemented by operational programs which includes periodic hydrostatic or pressure testing of pipe segments, instrument calibration, and when required, visual inspection and maintenance of piping, trench and instrument integrity.

A discussion of the radiological aspects of the system leakage is contained in DCD Section 11.1. Design and system features addressing RG 4.21 are captured in Section 12.3.1.3 of the DCD.

Monitoring the leakage from the primary side of the evaporator, the radiation monitor is attached to the downstream of the auxiliary steam drain pump. The high alarm of the monitor isolates the pump discharge line and steam supply line from main steam and trips the pump.

Group II components served by the system are shown below. These components are supplied auxiliary steam from the auxiliary boiler during plant startup, shutdown or regular inspections due to unavailable of the main steam.

- Turbine gland seal
- Deaerator seal
- Deaerator heating

The auxiliary steam for group II components are collected to the turbine cycle.

10.4.11.2.2 Component Description

Auxiliary steam drain tank

The auxiliary steam drain tank collects the condensed water from group I components.

Auxiliary steam drain tank pump

The auxiliary steam drain pump transfers the condensed water from the group I components to the condenser during plant normal operation or to the auxiliary boiler in situations in which main steam is not available.
The pumps are actuated by the high water level signal of the tank, and then stopped by the low signal. The pumps are also tripped by the auxiliary steam drain monitor high alarm. Two pumps are used by the system, another one is a spare.

MSIV operation and drain pot operation considers steam line water entrainment effects. Before opening the MSIV during plant start up, main steam piping down stream of the MSIV is warmed gradually by opening the MSBIV, hence, water slug formation in the condensate is prevented. The automatic power operated valve attached parallel to the steam trap opens when the drain pot high level switch activates, and the high level alarm is annunciated to MCR to give warning to the operator.

The Combined License Applicant is to provide operating and maintenance procedures including adequate precautions to prevent water (steam) hammer, relief valve discharge loads and water entrainment effects in accordance with NUREG-0927 and a milestone schedule for implementation of the procedure. The procedures should address:

- Prevention of rapid valve motion
- Introduction of voids into water-filled lines and components
- Proper filling and venting of water-filled lines and components
- Introduction of steam or heated water that can flash into water-filled lines and components
- Introduction of water into steam-filled lines or components
- Proper warmup of steam-filled lines
- Proper drainage of steam-filled lines
- The effects of valve alignments on line conditions.

10.3.3 Safety Evaluation

- Each main steam line is provided with MSSVs and MSRVs to automatically remove stored energy and to limit the pressure in the line.

- Each line is provided with a MSDV for controlled removal of reactor decay heat (in conjunction with the EFWS) during safe shutdown after plant transient and accident conditions.

- Redundant power supplies are provided to operate MSIVs and MSBIVs for containment isolation.

- Branch lines located on the safety-related portion of the main steam lines contain normally closed valves or power operated valves which are closed remotely when required.

- Radioactive contamination of the MSS can occur by a primary side to secondary side leak in the SG. ~~Under normal operating conditions, there are no significant amount of radioactivity in the MSS.~~ **<u>The main steam can become contaminated due to tritium diffusion through SG tubes even without primary-to-secondary leakage. A discussion of the radiological aspects of primary-to-secondary system leakage and conditions for operation is contained in Chapter 11.</u>** Additionally, the MSIVs provide controls for reducing releases by isolating the affected main steam line following a steam generator tube rupture (SGTR). In-line radiation monitors on each steam line, condenser vacuum pump exhaust line radiation monitor, GSS exhaust fan discharge line radiation monitor and the SG blowdown line radiation monitor facilitate leak detection.

- The safety-related portions of the MSS are located in the containment and the main steam/feedwater piping area of the reactor building. These buildings are designed to withstand the effects of earthquakes, tornadoes, hurricanes, floods, external missiles and other natural phenomena. Sections 3.3, 3.4, 3.5, 3.7 and 3.8 describe the bases of the structural design of these buildings.

(8) SG blowdown sample coolers

These coolers are provided to cool the sample line water to approximately 113°F. Component cooling water (CCW) is used for cooling the sample line water. Four sample coolers, one for each SG blowdown line is provided. Each cooler is sized for 100 percent capacity.

10.4.8.3 Safety Evaluation

- Redundant power operated isolation valves provided in each blowdown line isolate safety and non safety-related portions of the steam generator blowdown system. This preserves the secondary side water inventory to remove sensible and decay heat from the reactor coolant system.

- The SGBDS's safety-related functions are accomplished by redundant means. A single, active component failure within the safety-related portion of the system does not compromise safety function of the system. Power is supplied by the Class 1E power system as described in Chapter 8.

- Radioactive contamination of the SGBDS can occur by a primary to secondary leakage in the steam generator. ~~Under normal operating conditions, there is no significant amount of radioactivity in the steam generator blowdown.~~ **The SGBDS can become contaminated due to tritium diffusion through SG tubes even without primary-to-secondary leakage. A discussion of the radiological aspects of primary-to-secondary system leakage and conditions for operation is contained in Chapter 11.** The isolation valve(s) in each blowdown line provides controls for reducing releases by isolating the affected steam generator blowdown line following a steam generator tube rupture. An inline radiation monitor on the common line from the steam generator blowdown sample lines, facilitate leak detection.

- The safety-related portions of the SGBDS are located in the containment and the main steam/feedwater piping area. These buildings are designed to withstand the effects of earthquakes, tornadoes, hurricanes, floods, external missiles and other appropriate natural phenomena. Sections 3.3, 3.4, 3.5, 3.7 and 3.8 describe the bases of the structural design of these buildings. The safety-related portion of the SGBDS is designed to remain functional during and after a safe-shutdown earthquake.

- The safety-related components of the SGBDS are qualified to function in normal and accident environmental conditions. The environmental qualification program is described in Section 3.11.

- Section 3.2 provides quality group classification, design and fabrication codes, seismic category applicable to the SGBDS.

- Failure modes and effects analysis, as listed in Table 10.3.3-1, concludes that no single failure coincident with loss of offsite power compromises system's safety functions.

- High and moderate energy pipe break locations and its effects are discussed in Section 3.6.

λ = Decay constant (s-1)

BO = Initial boron concentration (ppm)

β = Boron dilution rate (ppm/s)

DF = Decontamination factor of demineralizer

QL = Reactor coolant letdown flow rate (g/s)

f = Fraction yield of radioactive decay product

FS = Stripping fraction of VCT

Subscript "p" refers to the parent nuclide.

Subscript "d" refers to the daughter nuclide.

The results of the calculations are listed in Table 11.1-2. The operation time indicated in Table 11.1-1 is the designed maximum time (the planned cycle duration is up to 24 months), therefore the activities tabulated are the values represent the maximum concentration, which is expected to occur during the equilibrium fuel cycle.

11.1.1.2 Corrosion Products

The activities of corrosion products are determined based on the existing plant data and are independent of the fuel defect level; these are given in Table 11.1-2.

11.1.1.3 Tritium

Tritium (H-3) is produced within the reactor coolant through the activation of soluble boron and soluble lithium contained within the reactor coolant. The presence of burnable neutron absorbers is another source of H-3 within the reactor core. The major source of H-3 is a fission product in the fuel (ternary fission) which can enter the reactor coolant system through the fuel cladding. Total H-3 production from both fission and activation is presented in Table 11.1-3. Within the coolant system, H-3 principally exists in combination with hydrogen in the tritiated oxide form. In this form, H-3 can not be easily removed from the coolant system. Therefore, it can not be effectively treated through cleanup processes. The activity of H-3 in the secondary side water and steam is entirely controlled by the loss of water from the reactor coolant system through primary-to-secondary leakage. A typical activity of H-3 in the reactor coolant is 1 μCi/g, as indicated in Table 11.1-9. A typical activity of H-3 common to the secondary side water and steam is 0.001 μCi/g, as stated in Table 11.1-9. **This activity is calculated based on a primary-to-secondary leakage rate of 75 lb/day with a moderate amount of condensate recycle. At higher primary-to-secondary leakage rate, up to and including 150 gallons per day, and with full recycle of condensate, tritium concentration is progressively higher, approaching reactor coolant concentration.**

11.1.1.4 Carbon-14

C-14 is produced through the activation of constituents such as N-14 and O-17 within the coolant. The activity of C-14 is less than 0.05 μCi/g based on 260 GBq/GWe/a

Table 11.1-9 Realistic Source Terms[a] (Sheets 1 of 2)

Noble Gases		
Nuclide	Reactor Coolant Activity (μ Ci/g)	SG Steam Activity (μ Ci/g)
Kr-85m	1.8E-02	2.8E-09
Kr-85	2.8E-01	4.3E-08
Kr-87	1.9E-02	8.3E-09
Kr-88	2.0E-02	3.1E-09
Xe-131m	6.8E-01	1.0E-07
Xe-133m	7.4E-02	1.2E-08
Xe-133	2.9E-02	4.5E-09
Xe-135m	1.4E-01	2.2E-08
Xe-135	7.4E-02	1.1E-08
Xe-137	3.8E-02	5.9E-08
Xe-138	6.8E-02	1.1E-08

Halogens			
Nuclide	Reactor Coolant Activity (μ Ci/g)	SG Water Activity (μ Ci/g)	SG Steam Activity[1] (μ Ci/g)
Br-84[2]	-	-	-
I-131	1.1E-03	2.3E-08	2.3E-10
I-132	5.6E-02	5.6E-07	5.6E-09
I-133	1.7E-02	3.1E-07	3.1E-09
I-134	1.0E-01	5.5E-07	5.5E-09
I-135	4.3E-02	6.6E-07	6.5E-09

Notes:

1. These values are calculated using the equation:
   SG Steam Activity = SG Water Activity x Partition Factor (see ANSI/ANS-18.1-1999 Table 9)
2. Because this nuclide's release is less than 1.0E-05 Ci/yr, these activities are not output by the PWR-GALE Code

Rubidium, Cesium			
Nuclide	Reactor Coolant Activity (μ Ci/g)	SG Water Activity (μ Ci/g)	SG Steam Activity (μ Ci/g)
Rb-88	2.1E-01	4.6E-07	2.3E-09
Cs-134	2.1E-05	4.2E-10	2.1E-12
Cs-136	5.1E-04	1.0E-08	5.2E-11
Cs-137, Ba-137m[1]	3.0E-05	6.2E-10	3.1E-12

Notes:

1. These nuclides are in secular equilibrium.

a. <u>SG Water and Steam Activities are based on a primary-to-secondary leakage rate of 75 lb/day and resulting secondary coolant activity as described in ANSI/ANS-18.1-1999</u>

Table 11.1-9 Realistic Source Terms[a] (Sheets 2 of 2)

Tritium			
Nuclide	Reactor Coolant Activity (μ Ci/g)	SG Water Activity (μ Ci/g)	SG Steam Activity (μ Ci/g)
H-3	1	1.0E-03	1.0E-03

Miscellaneous Nuclides			
Nuclide	Reactor Coolant Activity (μ Ci/g)	SG Water Activity (μ Ci/g)	SG Steam Activity (μ Ci/g)
Na-24	3.2E-02	5.6E-07	2.8E-09
Cr-51	1.7E-03	3.5E-08	1.7E-10
Mn-54	8.6E-04	1.7E-08	8.6E-11
Fe-55	6.5E-04	1.3E-08	6.5E-11
Fe-59	1.6E-04	3.2E-09	1.6E-11
Co-58	2.5E-03	5.1E-08	2.5E-10
Co-60	2.9E-04	5.8E-09	2.9E-11
Zn-65	2.7E-04	5.6E-09	2.6E-11
Sr-89	7.6E-05	1.5E-09	7.6E-12
Sr-90	6.5E-06	1.3E-10	6.5E-13
Sr-91	7.1E-04	1.2E-08	5.8E-11
Y-91m	4.8E-04	2.5E-09	1.2E-11
Y-91	2.8E-06	5.6E-11	2.8E-13
Y-93	3.1E-03	4.9E-08	2.4E-11
Zr-95	2.1E-04	4.3E-09	2.1E-11
Nb-95	1.5E-04	2.9E-09	1.5E-11
Mo-99	3.7E-03	7.2E-08	3.6E-10
Tc-99m	3.8E-03	5.2E-08	2.6E-10
Ru-103	4.1E-03	8.3E-08	4.1E-10
Ru-106	4.8E-02	9.8E-07	4.9E-09
Ag-110m	7.0E-04	1.4E-08	7.1E-11
Te-129m	1.0E-04	2.1E-09	1.0E-11
Te-129	2.4E-02	1.6E-07	8.0E-10
Te-131m	9.3E-04	1.7E-08	8.7E-11
Te-131	8.2E-03	2.4E-08	1.2E-10
Te-132	9.7E-04	1.9E-08	9.5E-11
Ba-140	7.1E-03	1.4E-07	7.1E-10
La-140	1.5E-02	2.9E-07	1.4E-09
Ce-141	8.1E-05	1.6E-09	8.2E-12
Ce-143	1.7E-03	3.2E-08	1.6E-10
Ce-144	2.2E-03	4.2E-08	2.1E-10
W-187	1.6E-03	2.9E-08	1.5E-10
Np-239	1.3E-03	2.5E-08	1.2E-10

a. <u>SG Water and Steam Activities are based on a primary-to-secondary leakage rate of 75 lb/day and resulting secondary coolant activity as described in ANSI/ANS-18.1-1999</u>