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April 25, 2012

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-12103

Subject: **MHI's Response to US-APWR DCD RAI No. 895-6172 Revision 3 (SRP 12.03-12.04)**

Reference:
1) "REQUEST FOR ADDITIONAL INFORMATION 895-6172 REVISION 3, SRP Section: 12.03-12.04 – Radiation Protection Design Features – Application Sections: 12, dated January 27,2012.
2) "MHI's Amended Responses to US-APWR DCD RAI No. 524-4020 Rev. 1 and No. 532-4019 Rev. 1", UAP-HF-10253, dated September 15, 2010.
3) "Request for Additional Information No. 524-4020 Rev. 1, SRP Section: 12.03-12.04 – Radiation Protection Design Features, Application Section: 12.3," dated January 26, 2010.

With this letter, Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") the document entitled "Response to Request for Additional Information No. 895-6172 Revision 3".

Enclosed is the response to RAI questions contained within Reference 1.

This response provides additional details about the containment racks installed inside the refueling cavity. Note that a small portion of the response to RAI 524-4019 Question 12.03-12.04-35 Items 2 & 4 related to where to place fuel in transient during a drain down event is superseded by the information included in the response to Question 12.03-12.04-47 in the enclosed document.

The enclosed document is being submitted in two versions. One version (Enclosure 1) includes certain information, designated pursuant to the Commission guidance as sensitive unclassified non-safeguards information, referred to as security-related information ("SRI"), that is to be withheld from public disclosure under 10 C.F.R. § 2.390. The information that is SRI is identified by brackets. The second version (Enclosure 2) omits the SRI and is suitable for public disclosure. In the public version, the SRI is replaced by the designation "[Security-Related Information - Withheld Under 10 CFR 2.390]".

Please contact Mr. Joseph Tapia, General Manager of Licensing Department, Mitsubishi Nuclear Energy Systems, Inc. if the NRC has questions concerning any aspect of the submittals. His contact information is below.

D081
NRO

Sincerely,

Y. Ogata

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

Enclosure:

1. Response to Request for Additional Information No. 895-6172 Revision 3 (SRI included version)

2. Response to Request for Additional Information No. 895-6172 Revision 3 (SRI excluded version)

CC: J. A. Ciocco
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Enclosure 2

UAP-HF-12103
Docket No. 52-021

Response to Request for Additional Information No. 895-6172
Revision 3

April 2012
(Security-Related Information Excluded)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

04/25/2012

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 895-6172 REVISION 3**
SRP SECTION:	**12.03-12.04 – RADIATION PROTECTION DESIGN FEATURES**
APPLICATION SECTION:	**DCD Sections 12**
DATE OF RAI ISSUE:	**01/27/2012**

QUESTION NO. : 12.03-12.04-40

The applicant's response to RAI 524-4020 Revision 1, dated 14 September 2010; Question 12.03-12.03-35 Item 1 stated that the Refueling Cavity has two fuel containment racks that are capable of temporarily storing a total of six fuel bundles in the Refueling Cavity. In their response, the applicant referred to Figure I "Locations of the Containment Racks" that was provided as part of the RAI response.

However, these racks are not depicted in any of the following US-APWR DCD Tier 2 Revision 2 figures:

Figure 3.8.3-6 "Interior Compartments Wall Layout and Configuration"
Figure 5.1-3 "Reactor Coolant System Loop Layout"
Figure 5.1-4 "Reactor Coolant System-Elevation"
Figure 9.1.2-2 "Spent Fuel Rack Array"
Figure 9.1.4-2 "Section View of Light Load Handling System"
Figure 12.3-1 "Radiation Zones for Normal Operation/Shutdown"

In addition, the involved subsections of the US-APWR DCD Tier 2 Revision 2 do not contain a description of or state the purpose of these racks, including:

- Subsection 3.8.3.1.7 "Refueling Cavity," which does not discuss temporary storage of fuel in the Refueling Cavity,
- Subsection 1.2.1.5.4.3 "Fuel Storage and Handling System," which states that spent fuel is stored in the spent fuel pit, without any mention of the temporary fuel storage racks in the Refueling Cavity,
- Subsection 1.2.1.7.2.1 "Reactor Building (R/B)," which describes the Refueling Cavity and fuel storage and handling area, without mentioning the temporary fuel storage racks in the Refueling Cavity,
- Table 1.9.2-9 "US-APWR Conformance with Standard Review Plan Chapter 9 Auxiliary Systems," which does not contain any discussion of the regulatory guidance applicable to the temporary storage of fuel in the Refueling Cavity,
- Subsection 9.1.2.2.2 "Spent fuel storage," which does not describe the use of temporary storage of fuel in the Refueling Cavity,

- Subsection 12.2.1.1.10 "Miscellaneous Sources," 12.2.1.2 "Sources for Shutdown," and 12.2.1.2.3 "Spent Fuel," do not discuss temporary storage of fuel in the Refueling Cavity.

Title 10 of the Code of Federal Regulations (10 CFR), Part 50 "Domestic Licensing of Production and Utilization Facilities" Appendix A "General Design Criteria for Nuclear Power Plants" (GDC) 61 "Fuel storage and handling and radioactivity control," requires that the fuel storage system be designed for adequate safety under anticipated operating and accident conditions. Based on the guidance contained in Standard Review Plan (SRP) subsection 9.1.2 "New and Spent Fuel Storage," the staff reviews the information on the facility design criteria, system description, and layout drawings for areas containing new and spent fuel. The guidance in SRP Section 12.2 "Radiation Sources," states that the descriptions of sources requiring shielding and special storage locations, including plan scale drawings, should be identified.

Please revise and update the US-APWR DCD to describe the temporary fuel storage racks located in the Refueling Cavity, including the purpose, drawings depicting the location, physical dimensions and elevations, the source terms and the regulatory basis, as noted in the subsections above, or provide the specific alternative approaches used and the associated justification.

ANSWER:

The US-APWR design includes two containment racks capable of temporarily storing up to six fuel assemblies in the Refueling Cavity. Three new technical topical reports for the Containment Racks will be issued in March 2013. At that time, DCD Subsections 9.1.1.3.4, 9.1.2.2.4 and 9.1.2.3.2 will be revised to reflect the technical reports design and analyses assumptions and results. The following DCD Subsections, Tables and Figures are added to the DCD to describe purpose, design criteria, location, physical description, etc., of the Containment Racks located in the Refueling Cavity that are utilized to facilitate fuel transfer operations, with accompanying changes reflecting compliance to applicable regulations and guidance:

1. Subsection 1.2.1.5.4.3 – The Fuel Storage and Handling System has been revised to include the Refueling Cavity and Containment Racks as part of this system.
2. Subsection 1.2.1.7.2.1 – The Reactor Building description has been revised to include the Containment Racks that are located on the north and west walls of the refueling cavity in the Pre-stressed Concrete Containment Vessel (PCCV).
3. Table 1.9.2-9, Sheet 1 of 31 – Standard Review Plan (SRP) Sections 9.1.1 and 9.1.2 on Criticality Safety of Fresh and Spent Fuel Storage and New and Spent Fuel Storage, respectively, has been revised to indicate compliance now includes the Containment Racks.
4. Subsection 3.1.6.2.1 – This section has been revised to show compliance with 10 CFR 50, Appendix A, General Design Criteria (GDC) 61 Fuel Storage and Handling and Radioactivity Control for the refueling cavity and fuel handling area in the PCCV with references to other applicable DCD sections and subsections for specific design details.
5. Subsection 3.1.6.3.1 – This section has been revised to show compliance with 10 CFR 50, Appendix A, GDC 62 Prevention of Criticality in Fuel Storage and Handling to indicate that the center-to-center spacing of the Containment Racks prevents criticality.
6. Subsection 3.2.1.1.1 – This section has been revised to specify that the Containment Racks are Seismic Category I.

7. Table 3.2-2, Sheet 45 of 56 – The table has been revised to show that the Containment Racks are Equipment Class 3, Quality Group C, comply with 10 CFR 50, Appendix B and Seismic Category I.
8. Subsection 3.8.3.1.7 – The description of the refueling cavity has been revised to include the Containment Racks with a pointer to Section 9.1 that now contains a detailed description of the Containment Racks.
9. Subsection 9.1.1.1 – Added the design description of the Containment Racks installed on the north and west walls of the refueling cavity, which references three new figures 9.1.2-3, 9.1.2-4, and 9.1.4-2 depicting the location, elevation, physical dimensions, and center-to-center spacing arrangement to preclude inadvertent criticality. Added that 10 CFR 50.68(b) Item (4) is applicable to the Containment Racks.
10. Subsection 9.1.1.2 – Added that the Containment Racks design is now described in Subsection 9.1.2.2.
11. Subsection 9.1.1.3 – Added the Containment Racks to the safety evaluation describing how the design prevents inadvertent criticality. Clarified that the new fuel storage racks and the Containment Racks design does not include boron in the stainless steel and is not credited in the criticality analysis.
12. Subsection 9.1.1.3.1 – Added that the Containment Racks comply with 10 CFR 50.68(b), Item (4).
13. Subsection 9.1.1.3.3 – Clarified that boron is not credited in the new fuel storage racks and Containment Racks criticality analysis.
14. Subsection 9.1.1.3.3 – Added to the criticality analysis section that the assumption used in the calculation for the Containment Racks was a finite rack cell array with the surrounding water reflectors.
15. Subsection 9.1.1.3.4 – Clarified that the referenced technical report was for the new fuel storage and spent fuel storage racks. Three new technical reports for the Containment Racks will be issued in March 2013. At that time, DCD Subsections 9.1.1.3.4, 9.1.2.2.4 and 9.1.2.3.2 will be revised to reflect the technical reports design and analyses assumptions and results.
16. Subsection 9.1.2.1 – Added the Containment Racks to the design bases for New and Spent Fuel Storage.
17. Subsection 9.1.2.2.3 – Added a new subsection with the design specifications for the Containment Racks.
18. Subsection 9.1.2.2.4 – Revised this subsection to reflect the addition of the new Subsection 9.1.2.2.3 on the Containment Racks design. This subsection will be revised in March 2013 when the technical reports are submitted as described in Item 15 previously.
19. Subsection 9.1.2.3.1 – Revised the New Fuel Racks Safety Evaluation section to include the Containment Racks since they have identical designs.
20. Subsection 9.1.2.3.3 – Revised the fuel assembly drop analysis to include drops over the Containment Racks.
21. Subsection 9.1.4.2.1.13 – Added new information on the water level channel operability before moving a fuel assembly and added that the reactor cavity water level alarm setpoint was determined using a water shielding depth that keeps personnel doses ALARA. Revised the subsection description of a draw down event that the fuel assembly in transit would be place in the nearest suitable safe storage location.
22. Subsection 9.1.4.2.2.2 – Added a description to the Reactor Refueling Operations that potential un-borated water sources are administratively closed. Added a description of how the refueling cavity, spent fuel pit, fuel inspection pit, and fuel transfer canal are interconnected and that the fuel transfer canal and these pits are opened during refueling; thus, maintaining water level between the refueling cavity,

fuel transfer canal and pit. Added a description of the low-level refueling cavity water alarm is set such that radiation levels for personnel are ALARA and water level is maintained for the necessary personnel shielding. Clarified that in the event that the refueling cavity low-level water alarm become inoperable for any reason that the spent fuel pit water level alarm will be used to take action should any leakage occur while fuel assemblies are in the Containment Racks located in the refueling cavity. Added that the refueling operations procedures will include the use of the Containment Racks, that during a drain down event, containment personnel will be notified immediately upon actuation of reactor cavity low-level water alarm RCS-LIA-011-N, and that should a drain down event occur, the refueling operations procedures will require the placement of the fuel assembly in the nearest safe location.

23. Table 9.1.2-2 – Added that the loads and load combinations for new and spent fuel racks, also includes the Containment Racks.
24. Table 9.1.2-3 – Added new light load drop conditions for the Containment Racks.
25. Figure 9.1.2-3 – New figure added depicting the location of the Containment Racks on the north and west walls of the refueling cavity which is in communication with fuel handling areas outside containment.
26. Figure 9.1.2-4 – New figure added depicting Plan View and physical dimensions for the Containment Racks.
27. Figure 9.1.4-2 – Amended light load handling system figure to include the elevation of the Containment Racks so that it can be readily seen that the elevation of the racks is well below the required depth of the refueling cavity water that is in communication with the SFP water.
28. Subsection 12.2.1.2.3 – Added to Spent Fuel Sources that the temporary storage of fuel assemblies in the refueling cavity Containment Racks is also a radiation source.
29. Subsection 12.3.2.2.2 – Added to the Containment Vessel Interior Shielding Design that the extensive shielding surrounds the refueling cavity and Containment Racks with a pointer to Section 9.1 for a detailed description of the Containment Racks.
30. Subsection 12.3.4 – Added that the Containment Racks are designed for all postulated normal and accident conditions and therefore do not need criticality monitors.
31. ITAAC 2.7.6.14 – Added new ITAAC for Containment Racks similar to what is already provided for the New and Spent Fuel Storage Racks.

The following Technical Specifications (TS) have been changed to reflect the temporary storage of fuel assemblies in the Containment Racks. A complete discussion of this addition is provided in response to Question 12.03-12.04-41:

1. Section 4.0 Design Features, new Subsection 4.3.1.3 – Added the criticality design features of the Containment Racks to the TS.

The following changes were made to Chapter 19 to reflect the Containment Racks. A more detailed discussion regarding these changes is provided in response to Questions 12.03-12.04-44 and 12.03-12.04-45:

1. Subsection 19.1.6.1 – The description of the Low-Power and Shutdown Operations PRA POS 6 was changed to state that this position includes when fuel is temporarily stored in the Containment Racks.
2. Table 19.1-81, Sheet 1 of 2 – Changed the Disposition of Plant Operating States for LPSD PRA to include POS 6 includes when fuel is temporarily stored in the Containment Racks.

Impact on DCD

Please see the attached mark ups to Tier 1 and Tier 2 Chapters 1, 3, 9, 12, 16 and 19.

Impact on R-COLA

There is no impact on the R-COLA.

Impact of S-COLA

There is no impact on the S-COLA.

Impact on PRA

There is no impact on the PRA.

Impact on Technical / Topical Report

A new technical report for the Containment Racks will be issued in March 2013.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

04/25/2012

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 895-6172 REVISION 3**
SRP SECTION:	**12.03-12.04 – RADIATION PROTECTION DESIGN FEATURES**
APPLICATION SECTION:	**DCD Sections 12**
DATE OF RAI ISSUE:	**01/27/2012**

QUESTION NO. : 12.03-12.04-41

The applicant's response to RAI 524-4020 Revision 1, dated 14 September 2010; Question 12.03-12.03-35 Item 1 stated that the Refueling Cavity has two fuel containment racks that are capable of temporarily storing a total of six fuel bundles in the Refueling Cavity.

Title 10 of the Code of Federal Regulations (10 CFR), Part 50 "Domestic Licensing of Production and Utilization Facilities" Appendix A "General Design Criteria for Nuclear Power Plants" (GDC) "Fuel storage and handling and radioactivity control," requires that the fuel storage system be designed for adequate safety under anticipated operating and accident conditions. GDC 62 "Prevention of criticality in fuel storage and handling," requires provisions to prevent unintended criticality. GDC 63 "Monitoring fuel and waste storage," requires systems to detect excessive radiation levels. The guidance in Standard Review Plan Section 9.1.4 "Light Load Handling System (Related to Refueling)" addresses handling of fuel and spent fuel, which, if dropped, mishandled, or damaged, could cause releases of radioactive materials or unacceptable personnel radiation exposures, and states that ANSI/ANS 57.1-1992 "Design Requirements for Light Water Reactor Fuel Handling Systems," provides guidance for meeting the requirements of GDC 61 and 62. ANSI/ANS-57.1-1992 in turn states that temporary storage locations shall be designed in accordance with ANSI/ANS 57.2-1983 "Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants."

US-APWR DCD Tier 2 Revision 2 Chapter 16, "Technical Specifications," (TS) Subsection 1.1 "Definitions," states that a MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 "Modes," with fuel in the reactor vessel. Based on this statement, Mode 6 "Refueling," is only applicable as long as there is fuel in the reactor vessel. TS Subsections 3.9.1 "Boron Concentration," TS 3.9.2 "Unborated Water Source Isolation Valves," 3.9.3 "Nuclear Instrumentation," 3.9.4 "Containment Penetrations," 3.9.5 "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level," which provides for reactivity controls, spent fuel cooling, separation of the

containment atmosphere from the environment, monitoring of neutron radiation from fuel and maintaining Refueling Cavity water level will be maintained when in Mode 6 "Refueling.

However, the Technical Specifications provided in Chapter 16 of the US-APWR DCD do not describe the required controls and limits when fuel is stored in the temporary fuel racks depicted in Figure I "Locations of the Containment Racks," when all fuel is out of the reactor vessel.

Please revise and update the US-APWR DCD Technical Specification section 3.9 "Refueling Operations," to provide reactivity controls, spent fuel cooling, separation of the containment atmosphere from the environment, monitoring of neutron radiation from fuel and maintaining Refueling Cavity water level when the plant is not in Mode 6 (i.e. all fuel is out of the reactor vessel) while fuel is still present in the Refueling Cavity, or provide the specific alternative approaches used and the associated justification.

ANSWER:

The use of the Containment Racks as a temporary storage location requires similar TS as required for the SFP. As described in response to Question 12.03-12.04-40, the TS have been revised to include a new Subsection 4.3.1.3 under Section 4.0 Design Features, Subsection 4.3 Fuel Storage.

New TS 4.3.1.3 includes the following criticality attributes:

The containment racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent,
b. $k_{eff} \leq 0.95$ if fully flooded with un-borated water, which includes an allowance for uncertainties as described in Subsection 9.1.1 of the DCD,
c. A nominal 16.9 inch center to center distance between fuel assemblies placed in containment racks.

This change is consistent with what has been previously approved on other new reactor designs with similar features and with existing DCD TS addressing the spent fuel storage pit. The DCD TS Section 3.9 on Refueling Operations does not need to be changed based upon the following:

- TS 3.9.1 on Boron Concentration is applicable in MODE 6 and requires a boron concentration to remain subcritical during refueling operations, and includes the refueling cavity and canal. Since this TS includes the refueling cavity (where the Containment Racks are installed) and refueling canal that are in communication, there is no need to change this TS because it applies to the refueling cavity and canal water before fuel assemblies would have been temporarily stored in the Containment Racks. Additionally, this TS would not be applicable if there was no fuel left in the reactor vessel. If the core had been off loaded and six assemblies were temporarily stored in the Containment Racks, the rack design is such that the center-to-center spacing precludes the necessity for adding boron for the fuel assemblies to remain subcritical. The Containment Rack design is based upon 5% enriched fuel assembly with $k_{eff} \leq 0.95$ if fully flooded with unborated water and center-to-center spacing that prevents criticality; these design requirements are now

described in new TS 4.3.1.3. Therefore, the boron concentration is irrelevant for maintaining the fuel assemblies in the Containment Racks subcritical.

- TS 3.9.2 on Unborated Water Source Isolation Valves is applicable in MODE 6 and would be applicable when the Containment Racks are in use as well. Before moving fuel assemblies, unborated water source isolation valves would have been previously secured, including before fuel assemblies would be temporarily stored in the Containment Racks. Further, the Containment Rack design is based upon unborated water with center-to-center spacing that would prevent criticality. Therefore, the introduction of unborated water to the fuel assemblies in the Containment Racks would not affect subcriticality, and as a result, this TS needs no modification.
- TS 3.9.3 on Nuclear Instrumentation is applicable in MODE 6 and applies without modification when fuel assemblies are temporarily stored in the Containment Racks. The Bases for this TS is that the audible count rate from the source range neutron flux monitors provides prompt and definite indication of any boron dilution. The count rate increase is proportional to the subcritical multiplication factor and allows operators to promptly recognize the initiation of a boron dilution event. This TS is not applicable to the Containment Racks since the design is based upon 5% enriched fuel assembly with $k_{eff} \leq 0.95$ if fully flooded with unborated water and center-to-center spacing that prevents criticality. Therefore, if a boron dilution event where to occur, there would be no affect on the criticality of the fuel assemblies temporarily stored in the Containment Racks since the design prevents criticality.
- TS 3.9.4 on Containment Penetrations applies during movement of irradiated fuel assemblies within containment. This does not specify where in containment an irradiated fuel assembly is being moved, only that it is being moved inside containment. This TS would apply to moving fuel assemblies to and from the Containment Racks as well, and identical ACTIONS would also apply. The purpose of the Containment Racks is to facilitate refueling operations by providing temporary storage (hours or days), and its use would be considered part of refueling operations (movement of irradiated fuel assemblies within containment). Therefore, this TS does not need to be changed because it also applies to moving fuel assemblies to and from the Containment Racks in containment.
- TS 3.9.5 on Residual Heat Removal (RHR) and High Water Level is applicable in MODE 6 with the water level ≥ 23 ft and does need to be modified due to the changes described previously. The Bases for this TS is to maintain adequate heat removal and boron mixing to prevent decay heat buildup and maintain subcriticality. If the core is off loaded from the reactor vessel, this TS is no longer applicable. Even if there were six fuel assemblies stored in the Containment Racks with the reactor vessel empty, there is sufficient heat load capability in the refueling cavity and fuel handling area water to prevent decay heat buildup. Boron mixing is not necessary for fuel assemblies stored in the Containment Racks since their design was based upon unborated water with center-to-center spacing that prevents criticality. Therefore, this TS does not require modification since the buildup of decay heat is not sufficient to require RHR cooling and the rack design is independent of requiring boron to maintain subcriticality.
- The Bases for TS 3.9.7 on Refueling Cavity Water Level is to require sufficient water level necessary to retain iodine fission product activity in the water in the event of a fuel handling accident. This TS is applicable during movement of irradiated fuel assemblies within containment whether or not fuel is in the vessel. Hence no change to this TS is necessary.

Impact on DCD

See the attached markup to new TS 4.3.1.3.

Impact on R-COLA

The R-COLA Part 4 Technical Specifications will be revised in the next revision to reflect these changes.

Impact on S-COLA

The S-COLA Part 4 Technical Specifications will be revised in the next revision to reflect these changes.

Impact on PRA

There is no impact on the PRA.

Impact on Technical / Topical Report

There is no impact on a technical or topical report.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

04/25/2012

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 895-6172 REVISION 3**
SRP SECTION:	**12.03-12.04 – RADIATION PROTECTION DESIGN FEATURES**
APPLICATION SECTION:	**DCD Sections 12**
DATE OF RAI ISSUE:	**01/27/2012**

QUESTION NO. : 12.03-12.04-42

The applicant's response to RAI 524-4020 Revision 1, dated 14 September 2010; Question 12.03-12.03-35 Item 1 stated that the Refueling Cavity has two fuel containment racks that are capable of temporarily storing a total of six fuel bundles in the Refueling Cavity.

Title 10 of the Code of Federal Regulations (10 CFR), Part 50 "Domestic Licensing of Production and Utilization Facilities" Appendix A "General Design Criteria for Nuclear Power Plants" (GDC) 61 "Fuel storage and handling and radioactivity control," requires that the fuel storage system be designed for adequate safety under anticipated operating and accident conditions. GDC 62 "Prevention of criticality in fuel storage and handling," requires provisions to prevent unintended criticality. Standard Review Plan (SRP) Subsection 9.1.1 "Criticality Safety of Fresh and Spent Fuel Storage and Handling," states that for Pressurized Water Reactors, where credit is taken for soluble boron in the pools containing fuel, reactivity design features are to be provided to ensure that K(eff) is less than 1.0 for all conditions, and that at the minimum allowable boron concentration that K(eff) be no greater than 0.95.

US-APWR DCD Tier 2 Revision 2 Subsection 9.1.1 "Criticality Safety of New and Spent Fuel Storage," states that under the new fuel assumption, the fuel assembly is assumed to have a maximum enrichment of 5 weight percent. Credit is taken for the neutron absorption in the rack structural stainless steel and a neutron poison, Metamic, built into the racks. US-APWR DCD Tier 2 Section 12.3 "Radiation Protection Design Features," states that the design of the fuel pool racks precludes criticality under all postulated normal and accident conditions.

However, the US-APWR DCD Subsections 9.1.1 and 12.3 do not describe the design features provided to preclude inadvertent criticality from the storage of 6 new fuel assemblies, containing the maximum enrichment of 5 weight percent, in the temporary fuel storage racks in the Refueling Cavity as described in Figure I "Locations of the Containment Racks."

Please revise and update the US-APWR DCD Subsections 9.1.1 and 12.3 to describe the design features provided to prevent inadvertent criticality of new fuel bundles temporarily stored in the Refueling Cavity temporary fuel storage racks, or provide the specific alternative approaches used and the associated justification.

ANSWER:

As described in the response to Question 12.03-12.04-40, Subsections 9.1.1, 9.1.1.2, 9.1.1.3, 9.1.1.3.1, 9.1.1.3.3 and 9.1.1.3.4 were revised to reflect that the Containment Racks temporary fuel storage design precludes an inadvertent criticality. Additionally, a new Subsection 9.1.2.2.3 on the Containment Racks design specifications was added. Subsection 9.1.1.3.3 was clarified by stating that the new fuel storage racks and Containment Racks do not utilize boron as a neutron absorber in the stainless steel nor is it credited in the criticality analysis for these racks. Metamic™ is used a neutron absorber in the spent fuel rack design and credited in the criticality analysis. Subsection 12.3.2.2.2 was changed to add the Containment Racks to the Containment Vessel Interior Shielding Design with a pointer back to Section 9.1 for a detailed description. Subsection 12.3.4 was changed to add that the Containment Racks are designed for all postulated normal and accident conditions and therefore do not need criticality monitors. Details of these changes are provided in response to Question 12.03-12.04-40 described previously.

Impact on DCD

See the attached markups to Subsections 9.1.1, 9.1.1.2, 9.1.1.3, 9.1.1.3.1, 9.1.1.3.3, 9.1.1.3.4, 9.1.2.2.3, 12.3.2.2.2 and 12.3.4.

Impact on R-COLA

There is no impact on the R-COLA.

Impact on S-COLA

There is no impact on the S-COLA.

Impact on PRA

There is no impact on the PRA.

Impact on Technical / Topical Report

There is no impact on a technical or topical report.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

04/25/2012

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 895-6172 REVISION 3**
SRP SECTION:	**12.03-12.04 – RADIATION PROTECTION DESIGN FEATURES**
APPLICATION SECTION:	**DCD Sections 12**
DATE OF RAI ISSUE:	**01/27/2012**

QUESTION NO. : 12.03-12.04-43

The applicant's response to RAI 524-4020 Revision 1, dated 14 September 2010; Question 12.03-12.03-35 Item 1 stated that the Refueling Cavity has two fuel containment racks that are capable of temporarily storing a total of six fuel bundles in the Refueling Cavity.

Title 10 of the Code of Federal Regulations (10 CFR), Part 50 "Domestic Licensing of Production and Utilization Facilities" Appendix A "General Design Criteria for Nuclear Power Plants" (GDC) 61 "Fuel storage and handling and radioactivity control," requires that the fuel storage system be designed for adequate safety under anticipated operating and accident conditions. The guidance in Regulatory Guide (RG) 1.183 "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Appendix B "Assumptions for Evaluating the Radiological Consequences of a Fuel Handling Accident," states that if depth of water above is less than 23 feet above spent fuel, the assumed decontamination factors will need to be determined on an individual basis. The guidance in RG-1.13 Revision 2 "Spent Fuel Storage Facility Design Basis," states that the minimum water depth above spent fuel should be 10 feet.

The applicant's response to RAI 524-4020, Question 12.03-12.03-35 Items 2 & 4 stated that the consequence analysis assumed that because fuel transfers were in progress, the isolation valve and weir gate between the Refueling Cavity and the spent fuel pool were open. With the spent fuel pool and the Refueling Cavity connected, the level of the water in the pools would only drop 5 feet, so the minimum depth above the reactor vessel flange would remain greater than 23 feet. However, if either the fuel transfer tube gate valve or the transfer canal weir gate were closed at the time of the event, the decrease in water level would be some value greater than 5 feet, increasing the dose rates around the Refueling Cavity to above the calculated values, and possibly reducing the water above the spent fuel to less than the depth provided in the guidance of RG 1.13 and RG 1.183.

Since the assumptions for the safe use of the temporary fuel storage racks in the Refueling Cavity depicted in Figure I "Locations of the Containment Racks," are not described in US-APWR DCD Tier 2 Subsections 9.1.2.2.2 "Spent fuel storage," 5.4.7.2.3.5 "Refueling,"

12.3.2.2.4 "Fuel Handling Area Shielding Design," or Technical Specification (TS) section 3.9 "Refueling Operations," the staff is unable to determine if the requirements of GDC 61 would be met.

Please revise and update the US-APWR DCD to describe the plant configuration needed to meet the requirements of GDC 61 when spent fuel is stored in the temporary fuel storage racks located in the Refueling Cavity, or provide the specific alternative approaches used and the associated justification.

ANSWER:

Compliance to GDC 61 for the refueling cavity and fuel handling area where the Containment Racks are located is now described in amended Subsection 3.1.6.2.1. A complete description of these changes is provided in response to Question 12.03-12.04-40.

Impact on DCD

See the attached mark up to Subsection 3.1.6.2.1.

Impact on R-COLA

There is no impact on the R-COLA.

Impact on S-COLA

There is no impact on the S-COLA.

Impact on PRA

There is no impact on the PRA.

Impact on Technical / Topical Report

There is no impact on a technical or topical report.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

04/25/2012

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 895-6172 REVISION 3**
SRP SECTION:	**12.03-12.04 – RADIATION PROTECTION DESIGN FEATURES**
APPLICATION SECTION:	**DCD Sections 12**
DATE OF RAI ISSUE:	**01/27/2012**

QUESTION NO. : 12.03-12.04-44

The applicant's response to RAI 524-4020 Revision 1, dated 14 September 2010; Question 12.03-12.03-35 Item 1 stated that the Refueling Cavity has two fuel containment racks that are capable of temporarily storing a total of six fuel bundles in the Refueling Cavity.

Title 10 of the Code of Federal Regulations (10 CFR), Part 50 "Domestic Licensing of Production and Utilization Facilities" Appendix A "General Design Criteria for Nuclear Power Plants" (GDC) 61 "Fuel storage and handling and radioactivity control," requires that the fuel storage system be designed for adequate safety under anticipated operating and accident conditions. GDC 63 "Monitoring fuel and waste storage," requires systems to ensure fuel safety. The guidance in RG-1.13 Revision 2 "Spent Fuel Storage Facility Design Basis," states that the minimum water depth above spent fuel should be 10 feet.

The applicant's response to RAI 524-4020, Question 12.03-12.03-35 Items 2 & 4 stated that MHI believes a rapid cavity drain down event is not considered feasible because the US-APWR permanent cavity seal (PCS) design prevents a seal cavity failure rapid drain down event and all cavity drain valves are administratively locked closed during fuel movement.

This RAI response appears to be inconsistent with the description of potential loss of coolant events during refueling conditions, as described in the US-APWR DCD Tier 2 Revision 2 Subsection 19.1.6.1 "Description of the Low-Power and Shutdown Operations PRA," which in the "Loss of coolant accident (LOCA)," and "LOCA [loss-of coolant accident] with failure of isolation and RCS makeup," which involve mispositioned or inadvertent opening of motor operated residual heat removal (RHR) system valves with the RHR pump running. US-APWR Table 5.4.7-2 "Equipment Design Parameters," states that the RHR pumps maximum flow rate is 3650 gpm, and Table 9.1.3-3 "Spent Fuel Pit Cooling and Purification System Component Design Parameters," states that the design flow rate of the spent fuel pool cooling pump is 3865 gpm. Since the assumptions for the maximum gravity driven drain down rate are based on a lower flow rate than could be achieved by a pump driven drain down rate, it is not clear to the staff that the projected dose calculation represents a conservative assessment of the projected dose rates from irradiated core components.

Please revise and update the US-APWR DCD to describe drain down dose rate calculation assumptions that are consistent with assumptions used in US-APWR DCD Chapter 19 "Probabilistic Risk Assessment and Severe Accident Evaluation," or provide the specific alternative approaches used and the associated justification.

ANSWER:

The drain down dose rate calculation assumptions are not consistent with the Chapter 19 PRA assumptions and scenario calculation because the two calculations are framed to evaluate different scenarios. The drain down event assumed in the dose rate calculation is conservatively based upon the inadvertent opening of an 8-inch drain valve, and evaluates the worst case dose workers might receive during a refueling cavity drawdown of 5 feet during a 30-minute period (reasonable time for operators to identify the origin of the leak) before the leak is detected and mitigative actions are taken to close the valve and restore water level. The PRA assumptions and scenario is to evaluate the probability of fuel exposure (i.e., core damage) without locating the leak within 30 minutes. In the PRA scenario, the event is mitigated by injecting water from the refueling water storage auxiliary tank into the spent fuel pit and allowing gravity feed to the refueling cavity to supplement reactor coolant system (RCS) makeup (see MHI's response to RAI 783-5855, Question 19-548 in letter UAP-HF-11274, dated August 24, 2011).

The drain down event is most likely to occur during refueling operation when the reactor coolant system (RCS) cavity is filled. The Chapter 19 low-power and shutdown (LPSD) probabilistic risk assessment (PRA) discusses the risk evaluation during mid-loop operation when the RCS water level is slightly above the top of the main coolant piping. The postulated plant condition of mid-loop operation considered in the LPSD PRA evaluation is inconsistent with initiating conditions of a drain down event in assessing a dose rate during refueling operations. Due to the difference in initiating conditions that exist for refueling operations to commence, the calculation assumption used in the drain down event is not applicable to the LPSD PRA.

Impact on DCD

There is no impact on the DCD.

Impact on R-COLA

There is no impact on the R-COLA.

Impact on S-COLA

There is no impact on the S-COLA.

Impact on PRA

There is no impact on the PRA.

Impact on Technical / Topical Report

There is no impact on a technical or topical report.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

04/25/2012

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 895-6172 REVISION 3**
SRP SECTION:	**12.03-12.04 – RADIATION PROTECTION DESIGN FEATURES**
APPLICATION SECTION:	**DCD Sections 12**
DATE OF RAI ISSUE:	**01/27/2012**

QUESTION NO. : 12.03-12.04-45

The applicant's response to RAI 524-4020 Revision 1, dated 14 September 2010; Question 12.03-12.03-35 Item 1 stated that the Refueling Cavity has two fuel containment racks that are capable of temporarily storing a total of six fuel bundles in the Refueling Cavity.

Title 10 of the Code of Federal Regulations (10 CFR), Part 50 "Domestic Licensing of Production and Utilization Facilities" Appendix A "General Design Criteria for Nuclear Power Plants" (GDC) 61 "Fuel storage and handling and radioactivity control," requires that the fuel storage system be designed for adequate safety under anticipated operating and accident conditions. The guidance in Regulatory Guide (RG) 1.183 "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Appendix B "Assumptions for Evaluating the Radiological Consequences of a Fuel Handling Accident," states that if depth of water above is less than 23 feet, the assumed decontamination factors will need to be determined on an individual basis. The guidance in RG-1.13 Revision 2 "Spent Fuel Storage Facility Design Basis," states that the minimum water depth above spent fuel should be 10 feet.

US-APWR DCD Tier 2 Revision 2 Subsection 19.1.6.1 "Description of the Low-Power and Shutdown Operations PRA," Plant Operation State (POS) 6 "No fuel in the core," states that for refueling and examination of fuel, fuel is transported from the reactor vessel (RV) to the spent fuel pool (SFP) during this POS. This state is excluded from the analysis because there is no fuel in the reactor. The end of POS 6 is defined as the time at which fuel is loaded into the reactor core. However, the risk analysis described in USAPWR DCD Tier 2 Revision 2 Chapter 19 "Probabilistic Risk Assessment and Severe Accident Evaluation," does not address the storage of fuel bundles in the Refueling Cavity temporary storage racks after all fuel has been removed from the reactor.

Please revise and update the US-APWR DCD the Chapter 19 "Probabilistic Risk Assessment and Severe Accident Evaluation," description of the POS 6 "No fuel in the core," to be consistent with the use of the temporary fuel storage racks located within the Refueling Cavity, or provide the specific alternative approaches used and the associated justification.

ANSWER:

The following changes were made to Chapter 19 to reflect the Containment Racks:

1. Subsection 19.1.6.1 – The description of the Low-Power and Shutdown Operations PRA POS 6 was changed to state that this position includes when fuel is temporarily stored in the Containment Racks.
2. Table 19.1-81, Sheet 1 of 2 – Changed the Disposition of Plant Operating States for LPSD PRA to include POS 6 includes when fuel is temporarily stored in the Containment Racks.

Impact on DCD

See the attached markups to Subsection 19.1.6.1 and Table 19.1-81, Sheet 1 of 2.

Impact on R-COLA

There is no impact on the R-COLA.

Impact on S-COLA

There is no impact on the S-COLA.

Impact on PRA

There is no impact on the PRA.

Impact on Technical / Topical Report

There is no impact on a technical or topical report.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

04/25/2012

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 895-6172 REVISION 3**
SRP SECTION:	**12.03-12.04 – RADIATION PROTECTION DESIGN FEATURES**
APPLICATION SECTION:	**DCD Sections 12**
DATE OF RAI ISSUE:	**01/27/2012**

QUESTION NO. : 12.03-12.04-46

Title 10 of the Code of Federal Regulations (10 CFR), Part 50 "Domestic Licensing of Production and Utilization Facilities" Appendix A "General Design Criteria for Nuclear Power Plants" (GDC) 61 "Fuel storage and handling and radioactivity control," requires that the fuel storage system be designed for adequate safety under anticipated operating and accident conditions. GDC 63 "Monitoring fuel and waste storage," requires systems to ensure fuel safety. The guidance contained in Regulatory Guide (RG) 1.13 "Spent Fuel Storage Facility Design Basis," states that reliable and frequently tested instruments, with local and remote alarms, should be provided for storage pool water level. The guidance contained in Standard Review Plan (SRP) section 9.1.2 "New and Spent Fuel Storage" for meeting GDC 63, refers to the guidance contained in RG 1.13.

The applicant's response to RAI 524-4020 Revision 1, dated 14 September 2010; Question 12.03-12.03-35 Item 1 stated that the Refueling Cavity has two fuel containment racks that are capable of temporarily storing a total of six fuel bundles in the Refueling Cavity. The applicant's response to Items 2 and 4 stated that a water level alarm from LIA 011–N, which is shown on DCD Rev. 2 Figure 5.1-2 "Reactor Coolant System Piping and Instrumentation Diagram," would detect a decrease in the Refueling Cavity water level. The operators were expected to be able to take action to isolate the source of the leakage within 30 minutes of receiving an alarm from LIA 011–N. However, the staff reviews of US-APWR DCD Chapters 5, 7, 9, 12, and 16 seem to indicate that there is no requirement to have this instrument in service while fuel is in the Refueling Cavity (RC), that there is no discussion of the required alarm setpoint for the minimum RC water level while fuel is in the RC and there is no discussion of what actions are required to be taken should the instrument not be in service. Also, since these instruments are normally provided for operation at the mid-loop reactor coolant system level, it is not clear to the staff that LIA 011-N alarm function is intended to be used for both cavity level control and mid-loop level control, or just cavity level control.

Please revise and update the US-APWR DCD to describe Refueling Cavity water level monitoring instrument, the operability requirements and the allowable set point used to assure adequate shielding of fuel in the RC, or provide the specific alternative approaches used and the associated justification.

ANSWER:

Level indicator alarm RCS-LIA-011-N (DCD Figure 5.1-2, Sheet 3 of 3) is wide range water level monitor used for refueling cavity level control only. Level indicator alarm RCS-LIA-012-N (DCD Figure 5.1.-2, Sheet 3 of 3) has a middle range water level for use during the reactor vessel head removal operation. Level indicator alarms RCS-LICA-014 and RCS-LICA-015 (DCD Figure 5.1-2, Sheet 3 of 3) have a narrow range water level for use during mid-loop operation. It should be noted that in addition to the alarm in the main control room, refueling cavity drain down may be determined by monitoring the level of the refueling water storage pit (RWSP), RWSAT or the volume control tank (VCT); however, these indications are not credited as a backup to RCS-LIA-011-N. The SFP water level alarm SFS-LIA-010-N and SFS-LIA-020-N is utilized as the backup.

In Subsection 9.1.4.2.1.13, new information was added on the water level channel operability before moving a fuel assembly. Additionally, Subsection 9.1.4.2.1.13 now specifies that the refueling cavity water level alarm setpoint was determined using a water shielding depth that keeps personnel doses ALARA. Finally, Subsection 9.1.4.2.1.13 now specifies that in the event that the refueling cavity water level alarm RCS-LIA-011-N becomes inoperable, the spent fuel pit water level alarm SFS-LIA-010-N and SFS-LIA-020-N will be utilized.

In Subsection 9.1.4.2.2.2, the Reactor Refueling Operations was changed to specify that each valve used to isolate un-borated water sources shall be secured in the closed position prior to commencing refueling operations. This section also describes how the refueling cavity, spent fuel pit, fuel inspection pit, and fuel transfer canal are interconnected and that the valves and weir's between the fuel transfer canal and these pits are opened during refueling; thus, maintaining water level between the refueling cavity, fuel transfer canal, spent fuel pit and refueling cavity. The low-level refueling cavity water alarm is set such that radiation levels for personnel are ALARA and water level is maintained for the necessary personnel shielding as fully described in Subsection 12.3.2.2.4. In the event that the refueling cavity low-level water alarm becomes inoperable for any reason, the spent fuel pit water level alarm will be used to take action should any leakage occur while fuel assemblies are in the Containment Racks located in the refueling cavity. Additionally, the refueling operations procedures will include the use of the Containment Racks, and that during a drain down event containment personnel will be notified immediately upon actuation of refueling cavity low-level water alarm RCS-LIA-011-N.

Impact on DCD

See the attached markups to Subsections 9.1.4.2.1.13 and 9.1.4.2.2.2.

Impact on R-COLA

There is no impact on the R-COLA.

Impact on S-COLA

There is no impact on the S-COLA.

Impact on PRA

There is no impact on the PRA.

Impact on Technical / Topical Report

There is no impact on a technical or topical report.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

04/25/2012

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 895-6172 REVISION 3**
SRP SECTION:	**12.03-12.04 – RADIATION PROTECTION DESIGN FEATURES**
APPLICATION SECTION:	**DCD Sections 12**
DATE OF RAI ISSUE:	**01/27/2012**

QUESTION NO. : 12.03-12.04-47

Title 10 of the Code of Federal Regulations (10 CFR), Part 50 "Domestic Licensing of Production and Utilization Facilities" Appendix A "General Design Criteria for Nuclear Power Plants" (GDC) 61 "Fuel storage and handling and radioactivity control," requires that the fuel storage system be designed for adequate safety under anticipated operating and accident conditions. GDC 63 "Monitoring fuel and waste storage," requires systems to ensure fuel safety.

The applicant's response to RAI 524-4020 Revision 1, dated 14 September 2010; Question 12.03-12.03-35 Item 1 stated that the Refueling Cavity has two fuel containment racks that are capable of temporarily storing a total of six fuel bundles in the Refueling Cavity. The applicant's response to Items 2 and 4 stated that a water level alarm from LIA 011–N, which is shown on DCD Rev. 2 Figure 5.1-2 "Reactor Coolant System Piping and Instrumentation Diagram," would detect a decrease in the Refueling Cavity water level, prompting the operators to place the fuel in transfer into a containment rack and evacuate from the vicinity of the Refueling Cavity. However, the staff reviews of US-APWR DCD Chapters 5, 7, 9, 12 and 16 have not identified any requirements to have one or more spaces in the temporary storage racks empty and available for this purpose during fuel movements.

Please revise and update the US-APWR DCD to describe Refueling Cavity temporary storage rack space availability requirements or an associated COL Item, or provide the specific alternative approaches used and the associated justification.

ANSWER:

DCD Subsection 9.1.4.2.2.2 has been revised to indicate that operating procedures will require the operators to place the in-transit fuel assembly in the nearest safe storage location. As a result, if the containment racks capacity has been filled, the operator will transfer the fuel assembly to the nearest safe storage location. As a result of the change to Subsection 9.1.4.2.2.2, requiring the operators to safely store a fuel assembly in the nearest

available location should a drain down event occur, it is unnecessary to require one or more spaces be available in the containment racks.

Impact on DCD

See the attached markups to Subsection 9.1.4.2.2.2.

Impact on R-COLA

There is no impact on the R-COLA.

Impact on S-COLA

There is no impact on the S-COLA.

Impact on PRA

There is no impact on the PRA.

Impact on Technical / Topical Report

There is no impact on a technical or topical report.

2.7.6.14 Containment Racks

DCD_12.03-12.04-40

2.7.6.14.1 Design Description

The purpose and function of the containment racks is to temporarily store new and irradiated fuel assemblies to facilitate re-fueling. Two containment racks, each with capacity for three fuel assemblies are located on the north and west walls of the refueling cavity in the PCCV.

The containment racks are located in the refueling cavity at an elevation whereby refueling cavity water can provide radiation shielding during refueling operations.

1. The functional arrangement of the containment racks is described in the Design Description of Subsection 2.7.6.14.1.

2. The containment racks are capable of maintaining fuel subcritical.

2.7.6.14.2 Inspections, Tests, Analyses and Acceptance Criteria

Table 2.7.6.14-1 describes the ITAAC for the containment racks.

DCD_12.03-12.04-40

Table 2.7.6.14-1 Containment Racks Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The functional arrangement of the containment racks is as described in the Design Description of Subsection 2.7.6.14.1.	1. Inspection of the as-built containment racks will be performed.	1. The as-built containment racks conform to the functional arrangement as described in the Design Description of Subsection 2.7.6.14.1.
2. The containment racks are capable of maintaining fuel subcritical.	2.i Inspections of the as-built containment racks will be performed.	2.i The as-built containment racks dimensions are consistent with the dimensions used in the containment rack criticality analysis.
	2.ii Inspections will be performed to verify that the materials of the as-built containment racks conform to the containment rack criticality analysis.	2.ii The materials of the as-built containment rack conform to the containment rack criticality analysis.

- Demineralizers and filters remove particulate and ionic impurities from the SFP water.
- The emergency power sources can supply electrical power to the SFP pumps, so that SFP cooling functions can be maintained during a LOOP.
- Water can be added to the system using the supply line from the demineralized water storage tank. In an emergency, replenishment of boric acid water can be accomplished using the supply line from the RWSP.
- The system is designed to maintain the water level of the SFP to prevent uncovering of stored fuel even if there is leakage due to failure of the piping.

The piping connected to the SFP is equipped with siphon breakers to prevent uncovering stored fuel in the event there is leakage in the system. During decay heat removal operation, SFP water flows from the SFP to the SFP pump suction, through the SFP cooler, transferring heat from the SFP water to the component cooling water, and returns to the SFP. A portion of the SFP water is diverted through the demineralizers and the filters in the purification part of the system to maintain water purity. During normal decay heat removal operation, one train can be used to purify the reactor cavity, the RWSP, and the RWSAT.

1.2.1.5.4.3 Fuel Storage and Handling System

The function of the fuel storage and handling system is to carry out fuel storage and handling safely and securely from the time the new fuel is brought into the power plant to the time the spent fuel is removed from the plant. The new fuel is stored in the new fuel pit in the R/B. After reactor shutdown, the spent fuel in the reactor is transferred to the spent fuel pit through the reactor cavity, refueling canal and fuel transfer tube, using the refueling machine, fuel handling machine, and relative fuel handling equipment.

All of the spent fuel transfer functions are carried out under boric acid water, which performs the functions of shielding and cooling. The spent fuel is stored in the spent fuel pit. After cooling is complete, the spent fuel is inserted into the spent fuel cask using the spent fuel cask handling crane, and then transported outside of the plant.

The major equipment of fuel storage and handling system are as follows.

- New Fuel Storage Pit, including new fuel storage racks DCD_12.03-12.04-40
- Spent Fuel Storage Pit, including spent fuel storage racks
- Cask Pit
- Cask Washdown Pit
- ~~Reactor~~Refueling Cavity, including containment racks and Refueling Canal DCD_12.03-12.04-40
- Refueling Machine

- Fuel storage and handling area
- Main steam and feed water area
- Safety-related electrical area

The containment facility is comprised of the PCCV and the annulus enclosing the containment penetration area, and provides an efficient leak-tight barrier and radiation protection under all postulated conditions, including a LOCA. The PCCV is a prestressed concrete structure designed to endure peak pressure under LOCA conditions. Access galleries are provided for periodic inspection and testing of circumferential and axial prestressing tendons.

For ease of access during operation, maintenance, repair, and refueling, the following accesses to the PCCV are also provided:

- A normal personnel airlock, located at floor level below the operating floor
- An equipment hatch and emergency airlock, located at operating floor level

The annulus is located adjacent to the PCCV and includes all penetration areas, to prevent the direct release of containment atmosphere to the environment through the containment penetrations. The pressure in the annulus is kept at a slightly negative level following accident conditions to control the release of radioactive materials to the environment.

The RWSP is located in the lowest part of the containment. The RWSP provides a continuous suction source for both the safety injection pumps and the CS/RHR pumps, thereby eliminating the switchover of suction source from the injection to the recirculation phase of accident recovery. The RWSP has four recirculation strainers on the floor, and the wall and floor of the RWSP are lined with stainless steel liner plates.

The reactor vessel is located at the center of the containment and is surrounded by a cylindrical concrete wall as a primary biological shield. There are four reactor coolant loops, each loop comprised of a steam generator, an RCP, and loop piping. Concrete walls surrounding the loops are provided as supporting media and as secondary biological shields. The pressurizer is located in its own compartment and is adjacent to the steam generators to minimize the length of the surge piping to the reactor coolant loop.

A refueling cavity with stainless steel liner is provided above the reactor vessel for refueling operations. Two containment racks are installed on the north and west walls of the refueling cavity. The fuel transfer tube connects this cavity to the fuel storage and handling area located outside the containment. The main steam and feedwater pipes that connect to the steam generators are routed within the containment with consideration of minimizing pipe run lengths, while providing sufficient flexibility to accommodate thermal expansion. DCD_12.03-12.04-40

Safety System Pumps and HX*s Area* - The safety system pumps (CS/RHR pumps and safety injection pumps), which require sufficient net positive suction head (NPSH) to draw

Table 1.9.2-9 US-APWR Conformance with Standard Review Plan Chapter 9 Auxiliary Systems (Sheet 1 of 31)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
9.1.1 Criticality Safety of Fresh and Spent Fuel Storage and Handling	1. The criteria for GDC 62 are specified in American National Standards Institute (ANSI)/American Nuclear Society (ANS) 57.1, ANSI/ANS 57.2, and ANSI/ANS 57.3, as they relate to the prevention of criticality accidents in fuel storage and handling.	Conformance with no exceptions identified. Note: This includes the containment racks.	9.1.1
9.1.2 New and Spent Fuel Storage	1. Acceptance for meeting the relevant aspect of GDC 2 is based on compliance with positions C.1 and C.2 of Regulatory Guide (RG) 1.13 and applicable portions of RG 1.29, and RG 1.117. For the spent fuel storage facility, additional guidance acceptable for meeting this criterion is found in American Nuclear Society (ANS) 57.2, 9.1.2-5 paragraphs 5.1.1, 5.1.3, 5.1.12.9, and 5.3.2. For the new fuel storage facility, additional guidance acceptable for meeting this criterion is found in ANS 57.3, paragraphs 6.2.1.3(2), 6.2.3.1, 6.3.1.1, 6.3.3.4, and 6.3.4.2. 2. Acceptance for meeting the relevant aspect of GDC 4 is based on positions C.2 and C.3 of RG 1.13, and RG 1.115 and 1.117. 3. GDC 5 is met by sharing the SSCs important to safety between the units in a manner that does not degrade the performance of their safety functions. 4. Acceptance for meeting the relevant aspect of GDC 61 for the spent fuel storage facility is based on compliance with positions C.4, C.6, C.10, C.11, and C.12 of RG 1.13 and the appropriate paragraphs of ANS 57.2. Acceptance for meeting this criterion for the new fuel storage facility is based on compliance with the appropriate paragraphs of ANS 57.3. Acceptance is also based on meeting the fuel storage capacity requirements noted in subsection III.1 of this SRP section. The following design considerations are evaluated:	Conformance with exceptions. Criterion 3 is not applicable for US-APWR design certification. (US-APWR is a single unit.) Note: This includes the containment racks.	9.1.2

DCD_12.03-12.04-40

DCD_12.03-12.04-40

Additionally, water may be added from several other sources, if required (Subsection 9.1.3). Adequate shielding is provided for the SFP and refueling cavity as described in Chapter 12. Radiation monitoring is provided as discussed in Chapters 11 and 12. DCD_12.03-12.04-40 DCD_12.03-12.04-43

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. Normal HVAC system for the refueling cavity area is provided by the Containment Ventilation System. ~~This~~These HVAC Systems ~~is~~are described in Chapter 9. DCD_12.03-12.04-40 DCD_12.03-12.04-43

The SFP cooling subsystem provides cooling to remove residual heat from the fuel stored in the SFP. The SFP cooling subsystem also provides cooling to the refueling cavity and fuel handling area water within containment during refueling operations when the SFP water is in communication with these areas. 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. DCD_12.03-12.04-40 DCD_12.03-12.04-43

The design of these systems meets the requirements of Criterion 61. The following sections further discuss fuel handling and storage systems inspection and testing, decay heat removal, purification, and prevention of reduction in coolant storage inventory: DCD_12.03-12.04-40 DCD_12.03-12.04-43

Section	Title
5.4.8	Reactor Water Cleanup
9.1.1	Fuel Storage and Handling
9.1.2	New and Spent Fuel Storage
9.1.3	Spent Fuel Pit Cooling and Purification System
9.1.4	Light Load Handling System (Related to Refueling)
9.1.5	Overheard Heavy Load Handling System
9.4.2	Spent Fuel Pool Area Ventilation System
9.4.3	Auxiliary Building Ventilation System
9.4.6	Containment Ventilation System
11.2	Liquid Waste Management System
11.4	Solid Waste Management System
12.3	Radiation Protection

3.1.6.3 Criterion 62 – Prevention of Criticality in Fuel Storage and Handling

Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

3.1.6.3.1 Discussion

Fuel storage and handling systems are provided to preclude accidental criticality for new and spent fuel. The restraints, interlocks, and geometrically safe physical arrangement provided for the safe handling and storage of new and spent fuel with respect to critically prevention are discussed and illustrated in Chapter 9.

As stated in Subsection 9.1.1, the new fuel storage racks, the spent fuel storage racks and containment racks are designed to have sufficient separation between adjacent fuel assemblies so the maximum k_{eff} under worst- case conditions is less than 1.0 without credit for the soluble boron, and less than 0.95 with partial credit taken for soluble boron. DCD_12.03-12.04-40

As also stated in Subsection 9.1.1, the new fuel storage racks are designed to have sufficient separation between adjacent fuel assemblies such that the maximum k_{eff} is less than 0.95 when flooded with unborated water, and less than 0.98 under optimum moderation conditions. DCD_12.03-12.04-40

New ~~and~~fuel storage racks, spent fuel storage racks, and containment racks are seismic category I components. DCD_12.03-12.04-40

The design of the spent fuel storage rack assembly is such that it is configurationally impossible to insert the spent fuel assemblies in other than prescribed locations, without physically modifying the rack, thereby preventing any possibility of accidental criticality.

Layout of the fuel handling area is such that a spent fuel cask cannot traverse the SFP.

See Chapter 9 for details.

3.1.6.4 Criterion 63 – Monitoring Fuel and Waste Storage

Appropriate systems shall be provided in the fuel storage and radioactive waste systems and associated handling areas (1) to detect conditions that may result in the loss of residual heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions.

3.1.6.4.1 Discussion

Instrumentation is provided to give indication and annunciation in the MCR of excessive temperature or low water level in the SFP. An area radiation monitor is provided in the fuel storage area for personnel protection and general surveillance. This area monitor alarms locally and in the MCR. Normally, the A/B HVAC System removes radioactivity from the atmosphere above the SFP and discharges it by way of the plant vent. The ventilation system is continuously monitored by gaseous radiation monitors. If radiation levels reach a predetermined point, an alarm is actuated in the MCR.

See Chapters 7, 9, and 12 for details.

design and classification for radioactive waste management SSCs. The use of the classification information and design criteria provided in the RG 1.143 (Reference 3.2-10) assures that components and structures used in RWMS are designed, constructed, installed, and tested in a manner that protects the health and safety of the public and plant operating personnel. Compliance with GDCs 2 and 61, as they relate to designing and constructing these SSCs to withstand earthquakes, and RG 1.143 (Reference 3.2-10), for seismic design and classification, provides assurance that SSCs containing radioactivity are properly classified and radiation exposures as a result of seismic events are as low as reasonably achievable.

RG 1.189 (Reference 3.2-11) is used as guidance to establish the design requirements of fire protection systems to meet the requirements of GDC 2 as it relates to designing these SSCs to withstand earthquakes. RG 1.189 is used to identify portions of fire protection SSCs requiring some level of seismic design consideration.

RG 1.189 (Reference 3.2-11), Positions 3.2.1, 6.1.1.2, and 7.1 are used to provide guidance for the proper seismic classification of fire protection systems. The use of this guidance assures that the fire protection systems for manual firefighting in areas containing safety-related equipment, containment penetrations, and reactor coolant pump (RCP) lube oil are properly classified and analyzed for SSE loads. Compliance with the above guidance assures that the safety-related SSCs required to function during an SSE are properly classified as seismic category I, and perform their safety functions.

3.2.1.1 Definitions

3.2.1.1.1 Seismic Category I

Seismic category I applies to safety-related SSCs (including their foundations and supports) that must remain functional and/or retain their pressure integrity in the event of an SSE.

This category includes SSCs designated as seismic category I in accordance with RG 1.29 (Reference 3.2-5). These SSCs are designed to withstand the effects of the SSE and maintain their structural integrity (including pressure integrity) and their specified design functions. The new and spent fuel pit structures, including fuel racks and refueling cavity structure including the containment racks, are designated seismic category I. Equipment Class 1, 2, or 3 components are designated seismic category I. DCD_12.03-12.04-40

Additionally, in accordance with RG 1.29 (Reference 3.2-5), systems, other than RWMS, that contain, or may contain, radioactive material whose postulated failure would result in potential offsite whole body doses that are more than the recommended limits, are classified as seismic category I.

Seismic category I SSCs are designed to withstand the effects of natural phenomena, including earthquakes, without jeopardizing the plant nuclear safety as discussed in Sections 3.7 and 3.10. The interaction of non-seismic category I structures with seismic category I structures is discussed in Subsection 3.7.2.8.

Seismic category I SSCs meet the QA requirements of 10 CFR 50, Appendix B (Reference 3.2-8).

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 39 of 56)

System and Components	Equipment Class	Location	Quality Group	10 CFR 50 Appendix B (Reference 3.2-8)	Codes and Standards(3)	Seismic Category(4)	Notes
Fuel inspection pit	3	R/B	C	YES	5	I	
Fuel transfer system	5	PCCV, R/B	N/A	N/A	5	II	
Suspension hoist and aux. hoist on spent fuel cask handling crane	5	R/B	N/A	N/A	5	II	
New fuel elevator	5	R/B	N/A	N/A	5	II	
Containment rack	~~6~~3	PCCV	~~N/A~~C	~~N/A~~YES	5	~~II~~I	DCD_12.03-12.04-40
New fuel assembly handling tool	5	R/B	N/A	N/A	5	NS	
Rod control cluster handling tool	5	R/B	N/A	N/A	5	NS	
Thimble plug handling tool	5	R/B	N/A	N/A	5	NS	
Burnable poison rod assembly handling tool	5	R/B	N/A	N/A	5	NS	
Control rod drive shaft handling tool	5	R/B	N/A	N/A	5	NS	
Permanent Cavity Seal	~~4~~3	PCCV	~~D~~C	~~N/A~~YES	5	~~II~~I	DCD_03.02.02-20
29. Containment System							
Containment vessel	2	PCCV	B	YES	2	I	
Equipment hatch	2	PCCV	B	YES	2	I	
Personnel hatch	2	PCCV	B	YES	2	I	

varies in elevation from 19 ft, 4 in. to 46 ft, 11 in. The top of the refueling cavity is 76 ft, 5 in. Additionally, containment racks are installed in the refueling cavity to temporarily store new or irradiated fuel assemblies. A more detailed description of the containment racks is provided in Section 9.1. DCD_12.03-12.04-40

The walls of the refueling cavity are formed by SC modules, which are lined with stainless steel over the 1/2-in. thick carbon steel plates, referred to as "clad steel." The ceiling and floor slabs are also lined with clad steel.

3.8.3.1.8 RWSP

The RWSP is located at the lowest part of the PCCV. The RWSP is formed by wall of SC modules using clad steel. A floor at elevation 3 ft, 7 in. is formed of clad steel in a layer of concrete that covers the containment liner and basemat. The ceiling is similarly lined with stainless steel. Subsection 6.2.1.1 provides a description of the RWSP layout and design features.

3.8.3.1.9 Interior Compartments

The containment internal structure includes several subcompartments designed to provide containment, radiation shielding, and protection of safety-related components. These compartments are formed by the secondary shield walls surrounding the primary loops from the SGs. They also protect the containment from postulated pipe ruptures inside the containment. These SC wall modules also form supports for intermediate floors and the operating deck at elevation 76 ft, 5 in. The walls are designed for load cases including earthquake and DBAs.

Subcompartments and/or rooms comprising the containment internal structure are summarized as follows:

- reactor cavity — EL. -9 ft, 2 in.
- containment drain sump room — EL. 9 ft, 6 in.
- letdown heat exchanger room — EL. 25 ft, 3 in.
- regenerative heat exchanger room — EL. 50 ft, 2 in.
- excess letdown heat exchanger room — EL. 50 ft, 2 in.

Labyrinths are provided beside the shield wall openings at several elevations for radiation protection, which consist of SC modules and reinforced concrete walls, floors, and ceilings.

Reinforced concrete slabs are used for the floor above the RWSP at elevation 25 ft, 3 in., the intermediate floor at elevation 50 ft, 2 in., and the operating floor at elevation 76 ft, 5 in. The floors are shown on the GA drawings in Chapter 1. The floor ~~is~~ at elevation 25 ft, 3 in.~~, and~~ is supported by the primary shield wall, the secondary shield wall, and the RWSP. MIC-03-03-00066 The floors at elevations 50 ft, 2 in. and 76 ft, 5 in. are supported by the secondary shield wall and the structural steel framing (beams and columns) arranged between the

9.0 AUXILIARY SYSTEMS

9.1 Fuel Storage and Handling

9.1.1 Criticality Safety of New and Spent Fuel Storage

9.1.1.1 Design Bases

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. New fuel is stored in low density racks installed in a dry new fuel storage pit. Spent fuel is stored in moderate density racks installed in a spent fuel pit (SFP) filled with borated water. Additionally, containment racks installed in the refueling cavity provide temporary storage for new or irradiated fuel assemblies during refueling operations. DCD_12.03-12.04-40 DCD_12.03-12.04-42

New fuel storage racks store 180 fuel assemblies, which corresponds to approximately one normal refueling batch plus an additional 50 locations. One normal refuel batch for the United States - Advanced Pressure Water Reactor (US-APWR) is one-half of a core. The center-to-center spacing between adjacent fuel assemblies is designed to be 16.9 in (as shown in Figure 9.1.1-1) to maintain subcriticality.

Spent fuel storage racks are capable to receive 900 fuel assemblies corresponding to the amount of spent fuel from ten years of operation at full power in case of a 24-month fuel cycle, plus one full-core discharge. The center-to-center spacing between adjacent fuel assemblies is designed to be 11.1 in (as shown in Figure 9.1.1-2) to maintain subcriticality.

Containment racks provide temporary storage for new or irradiated fuel assemblies to facilitate refueling operations. Two containment racks are located in the refueling cavity (as shown in Figure 9.1.2-3) that provide temporary storage for a total of 6 fuel assemblies with center-to-center spacing of 16.9 in. (as shown in Figure 9.1.2-4) to maintain subcriticality. DCD_12.03-12.04-40 DCD_12.03-12.04-42

The fuel storage and handling area is protected against natural phenomena. The robust concrete walls and ceiling surrounding the fuel storage and handling area is designed to withstand the loads and forces caused by earthquake, wind, tornados, floods and internal and external missiles.

New ~~and~~fuel storage racks, spent fuel storage racks, and containment racks are designed to maintain the required degree of subcriticality, and are evaluated as seismic category I structures. Equipment potentially damaging the stored fuel is designed to be prevented from collapsing and falling down on the structures in the event of a safe-shutdown earthquake (SSE). DCD_12.03-12.04-40 DCD_12.03-12.04-42

Criticality is precluded by adequate design of fuel handling and storage facilities and by administrative control procedures. The basic method of preventing criticality is the control of geometrically safe configurations. This is accomplished by providing geometrically safe spacing between assemblies to reduce neutron interaction. Credit for neutron absorber material is taken for the spent fuel storage rack and the spent fuel rack cells which

contain neutron absorber materials as fixed neutron poison. The design maintains K-effective (K_{eff}) at less than 1.0 for all normal and credible abnormal conditions. To provide additional margin, partial credit for soluble boron is taken into account for the evaluation. The fuel maximum reactivity assumption, worst case moderator density, and tolerances and uncertainties of the fuel and racks, are considered in order to maximize this calculated K_{eff} for normal and credible abnormal conditions.

Criticality analyses are performed in accordance with the following acceptance criteria and relevant requirements: General Design Criterion (GDC) 62 (Ref. 9.1.7-1), 10 CFR 50.68 (Ref. 9.1.7-2), NRC guide (Ref. 9.1.7-3), ANSI/ANS-8.17-2004 (Ref. 9.1.7-4), and relevant Standard Review Plan.

The 10 CFR 50.68 (b) item (2) and (3) for new fuel storage racks, ~~and~~ item (4) for spent fuel storage racks, and item (4) for containment racks are applied as the criticality safety design criteria. DCD_12.03-12.04-40 DCD_12.03-12.04-42

Criticality analysis codes are validated in accordance with ANSI/ANS-8.1-1998 (Ref. 9.1.7-5). The validation results are summarized in 4.3.3.2.

9.1.1.2 Facilities Description

The description of ~~new and spent fuel facilities is~~the new fuel storage racks, spent fuel storage racks, and containment racks are presented in Subsections 9.1.2.2. DCD_12.03-12.04-40 DCD_12.03-12.04-42

9.1.1.3 Safety Evaluation

Prevention of an inadvertent criticality is provided by adequate design of fuel handling and storage facilities and by administrative control procedures, considering the double contingency principle. The main methods for criticality control are (1) limiting the size of the array of fuel assemblies; and, (2) limiting the assembly neutron interaction by fixing the minimum separation and/or providing neutron poisons. In addition, rack cells are maintained in a safe geometry with no deformation in any design basis event. Flooding in the new fuel storage rack and boron dilution in the SFP and refueling cavity water are prevented or minimized. Fuel mishandling is prevented by the fuel handling procedures. DCD_12.03-12.04-40 DCD_12.03-12.04-42

For criticality safety design, the following analyses are performed to evaluate the degree of subcriticality and to verify compliance with the design criteria:

1. New fuel storage racks: The design is such that K_{eff} will not exceed 0.95 for flooded and 0.98 for optimum moderation conditions assuming single failure of sources of moderation and potential fire fighting activities. DCD_12.03-12.04-40 DCD_12.03-12.04-42

2. Spent fuel storage racks and containment racks: The minimum required soluble boron concentrations are evaluated for normal and accident conditions, pursuant to the criteria of 10 CFR 50.68 (b)(4). Postulated accident conditions are considered for dropping of a fuel assembly, abnormal location of a fuel assembly and rack movement in the event of seismic activity. Boron dilution events, if any, can be concluded to have no effect on criticality safety. DCD_12.03-12.04-40 DCD_12.03-12.04-42

Criticality analysis conditions are described below, including the design criteria, criticality analysis code with its validation for establishing code bias and bias uncertainty, and calculation model.

The guidance of RG 1.13 was considered in the design of the spent fuel storage facilities.

9.1.1.3.1 Design Criteria

The design criteria are pursuant to the 10 CFR 50.68 (b) item (2) and (3) for new fuel storage racks, ~~and~~ item (4) for spent fuel storage racks, and item (4) for containment racks. DCD_12.03-12.04-40 DCD_12.03-12.04-42

For new fuel storage racks, the maximum K_{eff} value including all biases and uncertainties must be less than or equal to 0.95 for the flooded condition with unborated water and less than or equal to 0.98 for optimum moderation, at a 95 percent probability, and 95 percent confidence level. Rack cells are assumed to be loaded with fuel of the maximum fuel assembly reactivity.

For spent fuel storage racks and containment racks, the maximum K_{eff} value, including all biases and uncertainties, must be less than or equal to 0.95 with partial credit for soluble boron credit and less than 1.0 with full density unborated water, at a 95 percent probability, and 95 percent confidence level. Rack cells are assumed to be loaded with fuel of the maximum fuel assembly reactivity. DCD_12.03-12.04-40 DCD_12.03-12.04-42

9.1.1.3.2 Analysis Code and Validation

Criticality safety analysis uses the three dimensional Monte Carlo code MCNP version 5.1 and the continuous-energy neutron library data ENDF/B-V, as summarized in Section 4.3.3.2.

A set of 120 critical benchmark experiments, from the "International Handbook of Evaluated Criticality Safety Benchmark Experiments" (Sep. 2006 Edition) (mentioned in Section 4.3.3.2), has been analyzed using the above code and library to demonstrate its applicability to criticality analysis and to establish the method bias and uncertainty.

The benchmark experiments cover a wide range of geometries, materials, and enrichments, and are considered adequate for qualifying methods for the analysis of storage facilities.

The analysis of the 120 critical experiments results in an average K_{eff} of 0.9971. Comparison with the measured values results in a method bias of 0.0029. The standard deviation for the set of experiments is 0.0030. For 120 samples and for a 95% probability at a 95% confidence level, the one-sided tolerance factor is 1.899.

9.1.1.3.3 Analysis Conditions

The following analysis conditions are assumed:

- Under the new fuel assumption, the fuel assembly is assumed to have a maximum enrichment of 5 weight percent which is pursuant to 10 CFR 50.68 (b) item (7).

- Fuel assembly fabrication tolerances are considered.

- Moderator is at the temperature (density) within the design limits that yields the largest reactivity. Full density of unborated water is assumed to be 62.43 lbm/ft^3. A moderator density range of 0 to 100 percent of full density is considered for the new fuel storage rack.

- Credit is taken for the neutron absorption in the rack structural material and neutron poison, such as boron for the spent fuel storage racks. The new fuel storage rack cell ~~consists~~and containment rack cell consist of stainless steel without boron, and boron is not credited in the criticality analysis assumptions for these racks. ~~and the~~The spent fuel storage rack cell consists of stainless steel with boron. Metamic is selected as nertron absorber material in the spent fuel racks. The steel plate thickness and boron content are conservatively set to a minimum. Performance effectiveness of the neutron absorber materials in the racks is taken into consideration. DCD_12.03-12.04-40 DCD_12.03-12.04-42

- The rack cell array is either assumed to be infinite in the lateral direction or is assumed to be surrounded by a conservatively chosen reflector, whichever is appropriate for the design:

New fuel storage rack

- A finite rack cell array and the surrounding concrete reflectors are used in the calculations.

Containment Racks DCD_12.03-12.04-40 DCD_12.03-12.04-42

- A finite rack cell array with the surrounding water reflectors is used in the calculations.

Spent fuel storage rack

- Basically, an infinite rack array in the lateral direction is used in the calculations. However, in the sensitivity study for determining uncertainty, the analysis model depends on the type of tolerance.

- Uncertainties are appropriately determined either by using worst-case conditions or by performing sensitivity studies. The uncertainties considered are material composition, fabrication tolerances of the fuel and rack, and the fuel location within the rack cell, as follows:

  - Steel plate thickness and its boron content are directly set to minimum so as to maximize $K_{eff.}$

- Other uncertainties are considered less effective and independent and are therefore statistically combined with the analysis code bias uncertainty.

The criticality evaluation is performed in accordance with Section 5.1 of ANSI/ANS-8.17-2004. Section 5 describes the following relationships.

$$k_p \leq k_c - \Delta k_p - \Delta k_c - \Delta k_m,$$

If the various uncertainties are independent,

$$k_p \leq k_c - (\Delta k_p^2 + \Delta k_c^2)^{1/2} - \Delta k_m.$$

where:

k_p is the calculated K_{eff}

k_c is the mean K_{eff} derived from the code validation

Δk_p is an allowance; calculation, tolerances

Δk_c is a bias uncertainty derived from the code validation

Δk_m is an arbitrary margin to ensure the subcriticality of k_p.

9.1.1.3.4 Criticality analysis for new fuel storage racks and spent fuel storage racks

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DCD_12.03-12.04-42

Criticality analysis for new fuel storage racks, and spent fuel storage racks is provided in the technical report (Ref.9.1.7-6).

DCD_12.03-12.04-40
DCD_12.03-12.04-42

9.1.2 New and Spent Fuel Storage

9.1.2.1 Design Bases

Subsection 9.1.1.1 provides the design bases for the new and spent fuel storage facilities and containment racks, including quantities of fuel to be stored and the configuration of the storage facilities.

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Storage racks for new fuel are designed of austenitic stainless steel with consideration for corrosion resistance. New fuel pit criticality, including flooding with a low density worst case moderator, is discussed in detail in Subsection 9.1.1.

The new fuel is protected from a heavy load drop accident by the limitation of travel of the heavy load handling crane preventing it from traveling over the new fuel pit. The heavy load handling crane is described in detail in Subsection 9.1.5. Failure modes of the fuel handling machine are described in Subsection 9.1.4. Drain facilities are provided to prevent the new fuel pit from flooding. New fuel pit nuclear safety and criticality issues are discussed in Subsection 9.1.1.

(2) Neutron attenuation measurements are a precise instrumental method of chemical analysis for Boron-10 content using a nondestructive technique in which the percentage of thermal neutrons transmitted through the panel is measured and compared with predetermined calibration data. Boron-10 is the nuclide of principal interest since it is the isotope responsible for neutron absorption in the Metamic panel.

Changes in excess of either of these two criteria requires investigation and engineering evaluation, which may include early retrieval and measurement of one or more of the remaining coupons to provide corroborative evidence that the indicated changes are real. If the deviation is determined to be real, an engineering evaluation shall be performed to identify further testing or any corrective action that may be necessary.

The remaining measurement parameters serve a supporting role and should be examined for early indications of the potential onset of Metamic degradation that would suggest a need for further attention and possibly a change in measurement schedule. These include 1) visual or photographic evidence of unusual surface pitting, corrosion or edge deterioration, or 2) unaccountable weight loss in excess of the measurement accuracy.

Design of the spent fuel storage facility is in accordance with Regulatory Guide 1.13.

The SFP is also provided with an array of 12 storage spaces for damaged fuel assembly containers. These racks do not contain the neutron absorber and the center-to-center spacing of this array is 24 inches.

No overhead crane, except the light load fuel handling machine, pass over the SFP. The fuel handling machine is designed to withstand seismic category I loads to preclude its fall or collapse due to an SSE.

9.1.2.2.3 Containment Racks

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DCD_12.03-12.04-42

Two containment racks are installed in the refueling cavity on the north and west walls (Figure 9.1.2-3) to temporarily store six new or irradiated fuel assemblies. The containment rack design is identical to the new fuel storage racks. This design is considered conservative because the center-to-center spacing provides minimum separation to maintain a subcritical array. Additionally, surfaces that come into contact with the fuel assemblies are made of annealed austentic stainless steel, and are smooth (125AA) in accordance with the requirement of ANSI/ANS-57.2.

9.1.2.2.4 New Fuel Storage Rack and Spent Fuel Storage Rack Design

The fuel storage facilities are designed to meet the guidelines of ANS 57.2 (Ref. 9.1.7-7) and ANS 57.3 (Ref. 9.1.7-9). Structural design and stress analysis of the new and spent fuel storage racks are evaluated in accordance with the seismic category I requirements of Regulatory Guide 1.29.

The dynamic and stress analyses are performed and described in the technical report (Ref. 9.1.7-8). Loads and load combinations considered in the structural design and stress analysis are shown in Table 9.1.2-2 based on SRP Section 3.8.4, Appendix D.

Uplift force analysis is also performed for new and spent fuel racks design, and described in the technical report (Ref. 9.1.7-8). Each rack is evaluated for withstanding a maximum uplift force of 4,400 pounds based on the lifting capacity of the suspension hoist and the fuel handling machine. Structural analysis is performed to verify that resultant stress in the critical part of the rack is within acceptable stress limits and deformation of the rack array is limited to maintain a subcritical array.

Fuel assembly drop analysis is performed for each fuel rack to maintain a subcritical array. Drop weight is determined from the maximum weight handled for each rack and drop height is determined from the higher value of 2 ft or the design height for handling fuel above each rack. The analysis is also provided in the technical report (Ref. 9.1.7-8)

9.1.2.3 Safety Evaluation

9.1.2.3.1 New Fuel Storage Racks and Containment Racks

DCD_12.03-12.04-40

The new fuel rack and containment racks~~, being a~~ are seismic category I structures~~, is~~ designed to withstand normal and postulated dead loads, live loads, loads resulting from thermal effects, and loads caused by the SSE event.

The new fuel rack is located in the new fuel storage pit, which has a cover to protect the new fuel from debris. ~~No loads are required to~~Loads will not be carried over the new fuel storage pit while the cover is in place. The cover is designed such that it will not fall and damage the fuel or fuel rack during a seismic event. Administrative controls are utilized when the cover is removed for new fuel transfer operations to limit the potential for dropped object damage. DCD_12.03-12.04-40

The containment racks are located on the north and west walls of the refueling cavity within the PCCV. DCD_12.03-12.04-40

The racks are ~~is~~ also designed with adequate energy absorption capabilities to withstand the impact of a dropped fuel assembly from the maximum lift height of the suspension hoist of the spent fuel cask handling crane as discussed in Subsection 9.1.2.3.3. Handling equipment (spent fuel cask handling crane) capable of carrying loads heavier than fuel components is prevented from carrying heavy loads over the fuel storage area. The fuel storage rack can withstand an uplift force greater than or equal to the uplift capability of the suspension hoist of the spent fuel cask handling crane (4,400 lbs).

Materials used in rack construction are compatible with the storage pit and refueling cavity environments, and surfaces that come into contact with the fuel assemblies are made of annealed austenitic stainless steel. Structural materials are corrosion resistant and will not contaminate the fuel assemblies or pit or refueling cavity environments. DCD_12.03-12.04-40 DCD_12.03-12.04-40

The new fuel assemblies are stored dry. The rack structure is designed to maintain a safe geometric array for normal and postulated accident conditions. The rack structure maintains the required degree of subcriticality for normal and postulated accident conditions such as flooding with pure water and worst case moderator density.

A discussion of the methodology used in the criticality analysis is provided in Subsection 9.1.1.

9.1.2.3.2 Spent Fuel Racks

The racks, being seismic category I structures (described in Section 3.2), are designed to withstand normal and postulated dead loads, live loads, loads resulting from thermal effects, and loads caused by the SSE event.

The racks are designed with adequate energy absorption capabilities to withstand the impact of a dropped fuel assembly from the maximum lift height of the fuel handling machine as discussed in Subsection 9.1.2.3.3. Handling equipment such as the cask handling crane which is capable of carrying loads heavier than fuel components is prevented by design from carrying loads over the spent fuel storage area. The fuel storage racks can withstand an uplift force greater than or equal to the uplift capability of the fuel handling machine (4,400 lbs).

Materials used in rack construction are compatible with the storage pool environment, and surfaces that come into contact with the fuel assemblies are made of annealed austenitic stainless steel. Structural materials are corrosion resistant and will not contaminate the fuel assemblies or pool environment. Neutron absorber material used in the rack design has been qualified for the storage environment.

Design of the spent fuel storage facility is in accordance with Regulatory Guide 1.13. A discussion of the methodology used in the criticality analysis is provided in Subsection 9.1.1. The thermal-hydraulic analysis demonstrating the flow through the spent fuel rack is adequate for decay heat removal from the spent fuel assemblies during anticipated operating conditions is provided in the technical report (Ref. 9.1.7-26).

9.1.2.3.3 Fuel Assembly Drop Analysis

Each new and spent fuel storage rack and containment rack ~~are~~is evaluated for withstanding a postulated drop of a fuel assembly and its associated handling tool to maintain a subcritical array assuming the maximum weight handled on each rack and the maximum drop height as described in Table 9.1.2-3. DCD_12.03-12.04-40

9.1.3 Spent Fuel Pit Cooling and Purification System

The spent fuel pit cooling and purification system (SFPCS) performs the following functions:

- Cools the SFP water by removing the decay heat generated by spent fuel assemblies in the SFP
- Purifies and clarifies the SFP water
- Purifies the boric acid water for the refueling water storage pit (RWSP), the refueling cavity, and the refueling water storage auxiliary tank (RWSAT) in conjunction with the refueling water system (RWS)
- Transfers boric acid water to the fuel transfer canal, fuel inspection pit, and cask pit in conjunction with the refueling water system.

9.1.4.2.1.12 Control Rod Drive Shaft Handling Tool

The control rod drive shaft handling tool is used to latch and unlatch the control rod drive shaft from the rod control cluster. It is suspended from the auxiliary hoist of the refueling machine.

9.1.4.2.1.13 Permanent Cavity Seal

DCD_09.01.04-22

The Permanent Cavity Seal (PCS) has a function to maintain water level in the refueling cavity during refueling operation by sealing an annular gap between the reactor vessel flange and the refueling cavity floor.

The seal is made of a stainless steel structure and permanently attached to the vessel and the floor with bolts and welds. Appropriate sections of the ASME Code. or codes and standards recommended by manufacturers shall be applied in selection of material and manufacture of the seal.

Should a load, such as a fuel assembly, suspended from the polar crane or refueling machine, which are designed as single failure proof, be dropped on the seal, damage to the seal is prevented by a stainless steel guard plate (curing lid) which is installed over the PCS. Moreover. since the PCS and the guard plate are washed thoroughly with demineralized water after the draining of the refueling cavity water to remove extraneous materials such as sludge, these structures do not degrade over time.

Leakage detection systems are utilized for the PCS. Two leakage detection pipes are installed under the ring-shaped PCS directly opposite each other across the reactor vessel. Wherever around the seal leakage from the PCS should occur, the leakage water flows and accumulates into an annular space between a vertical cylindrical plate, which is attached to support ring, and the PCS, and eventually flows into one or both of the detection pipes. Once water flows into the leakage detector via the leakage detection pipe, the leak detection system provides an alarm signal to alert operators in the MCR and in the vicinity of the fuel handling system that an abnormal water level condition exists in the refueling cavity.

The refueling cavity water High and Low level is monitored by a refueling cavity water level indicator and an alarm. which are shown as "RCS-LIA 011-N" in Figure 5.1-2 (Sheet 3 of 3). This water level channel is operable before the fuel assembly is moved from or to the R/V. The low level alarm setpoint is determined using a water shielding depth necessary to keep personnel radiation dose ALARA in the fuel handling area and refueling cavity. In the event that the refueling cavity water level alarm RCS-LIA-011-N becomes inoperable, the spent fuel pit water level alarm SFS-LIA-010-N and SFS-LIA-020-N will be utilized. The water shielding depth requirement for the refueling cavity, and the resulting radiation dose limit in the Fuel Handling Area are described in Subsection 12.3.2.2.4.

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Although a rapid cavity drain-down event is unlikely, if such an event should occur, upon alarm the workers immediately place any fuel in transit in the nearest suitable safe storage location. Since the seal is visually inspected before filling the cavity, the possibility of a rapid cavity drain-down event at a flow rate of more than 1 gpm, resulting from a large crack, which would be detected through visual inspection, is excluded. Therefore,

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sufficient time will be available to place any fuel in transit in the nearest suitable safe storage location. before the refueling cavity water level drops below the minimum level necessary to maintain proper shielding. DCD_12.03-12.04-40 DCD_12.03-12.04-46

9.1.4.2.2 Fuel Handling Operations

9.1.4.2.2.1 New Fuel Receipt

New fuel is shipped to the site in a new fuel shipping container. The new fuel shipping container is received into the R/B by way of the refueling area truck access bay at elevation 3 ft - 7 in.

The new fuel shipping container is raised from the truck using the auxiliary hoist on the spent fuel cask handling crane through the access hatch in the refueling area floors at elevations 25 ft - 3 in and 76 ft - 5 in. Elevation 76 ft - 5 in is the operating level of the refueling area.

The new fuel container is set on the operating floor. Using the suspension hoist on the spent fuel cask handling crane, new fuel is removed from the shipping container and stored in the new fuel storage pit. During this operation, the new fuel assemblies are suspended using a short fuel handling tool to permit surface inspection prior to being placed into a new fuel storage rack.

A new fuel assembly stored in the new fuel storage racks is transferred to the spent fuel pit to prepare for refueling.

A new fuel assembly stored in the new fuel racks is lifted using the suspension hoist of the spent fuel cask handling crane, and transferred to the new fuel elevator located in the fuel inspection pit. The new fuel assembly is then lowered using the new fuel elevator for access by the fuel handling machine. The new fuel assembly is latched by the spent fuel assembly handling tool on the fuel handling machine, and is lifted using the fuel handling machine mast tube or auxiliary hoist and then transferred to the spent fuel pit for temporary storage in the spent fuel rack.

General arrangement figures for the US-APWR are presented in Subsection 1.2.1.7.

9.1.4.2.2.2 Reactor Refueling Operations

During refueling, the refueling cavity is filled with water transferred from the RWSP. If a leakage from the refueling cavity occurred, the water level drops and alarms the MCR. Upon alarm and MCR action, the water level will be recovered by transferring water from the RWSP, using the refueling water recirculation pump. A sufficient quantity of water remains in the RWSP after the refueling cavity is initially filled with water to maintain the water level of the refueling cavity. DCD_09.01.04-22

Reactor refueling operations are divided into four phases: preparation, reactor disassembly, fuel handling, and reactor assembly. Refueling operations are outlined below and performed in accordance with operating procedures defined in Subsection 13.5.2.

- Phase I - Preparation

The reactor is placed into cold shutdown mode as defined in the Technical Specifications, Chapter 16. The refueling water and reactor coolant are borated to assure the core remains approximately 5% below criticality during refueling operations based on the maximum reactivity of the fuel to be cycled through an US-APWR. Lines that are potential sources of unborated water are closed administratively. DCD_12.03-12.04-40

The water level in the refueling cavity and the spent fuel handling pit and interconnected pits is maintained at an elevation sufficient to keep radiation levels within personnel access limits when the fuel assemblies are being removed and transported from the core to the spent fuel racks in accordance with RG 1.13. The refueling cavity, spent fuel pit, fuel inspection pit, and fuel transfer canal are interconnected. The fuel transfer tube valve and the gates between the fuel transfer canal and these pits are opened during refueling operations. Thus, the water level of the cavity, canals and pits are kept at the same level. The level meter of the SFP acts as alternative measurement of the refueling cavity level during the transfer of fuel. DCD_12.03-12.04-40 DCD_12.03-12.04-46 and 47

The water level is maintained above low water level to keep radiation levels within personnel access limits when the fuel assemblies are being removed and transported from the core to the spent fuel racks in accordance with RG 1.13. The low water level alarm of the refueling cavity is set at the required water depth for radiation shielding described in Subsection 12.3.2.2.4. The radiation and environmental levels are monitored to assure levels do not exceed personnel access limits. DCD_12.03-12.04-46 and 47

Upon achieving safe radiation and environmental conditions, the LLHS system is tested and the refueling machine overload is verified to be within operable. This is accomplished by using the mockup fuel assembly nozzle attached to the floor of the refueling cavity.

- Phase II – Reactor Disassembly

The reactor vessel head assembly is prepared for refueling by disconnecting electrical cabling, seismic support tie rods, in-core instrumentation, and cooling duct work. The refueling cavity is prepared by:

- Closing and locking the reactor cavity drain line
- Removing the blind flange of the fuel transfer tube
- Verifying functionality of the reactor cavity lighting
- Verifying tools are in place and functional
- Verifying the fuel transfer system is functional

fuel inspection pit to perform underwater visual inspections before transferring to the spent fuel rack, or inspected after completion the refueling (during normal operation). This process is continued until the core is off loaded. SFP level is maintained at normal throughout the refueling process to assure adequate radiation protection for personnel.

- The rod control clusters, the thimble plugs, and the burnable poison rod assemblies are shuffled in the SFP by using long handled tools on the fuel handling machine bridge.

- Irradiated and new fuel assemblies are individually lifted from a spent fuel rack by using the fuel handling machine, transferred to the up ender, and transferred to inside containment by reversing the core unloading process.

- Phase IV – Reactor Assembly

  The reactor assembly is accomplished by reversing the process described in Phase II – Reactor Disassembly.

Plant procedures contain measures to prevent and mitigate inadvertent reactor cavity drain-down events. Reactor refueling procedures require that valve positions of potential reator cavity drain paths are verified prior to filling the refueling cavity. Operating procedures direct operators to monitor control room indications for reactor cavity seal leakage during refueling operations. Maintenance procedures address periodic maintenance and inspection of the permanent cavity seal and other seals and plugs in accordance with vendor recommendations. Emergency response procedures provide direction to operators regarding the proper response to pool drain down events. During an inadvertent drain down event, containment personnel will be notified immediately upon actuation of refueling cavity low-level water alarm RCS-LIA-011-N. Fuel handling operating procedures will require that the operators place any fuel in transit in the nearest suitable safe storage location. Operating procedures are defined in Subsection 13.5.2. DCD_12.03-12.04-40 DCD_12.03-12.04-46 and 47

9.1.4.2.2.3 Spent Fuel Storage

The spent fuel assemblies are stored in the SFP until fission product activity is low enough to permit shipment from the site or to be placed in dry storage. Spent fuel storage and cooling is discussed in Subsections 9.1.2 and 9.1.3, respectively.

9.1.4.2.2.4 Spent Fuel Shipment

The procedure for the spent fuel shipment is as follows:

- The spent fuel cask is received into the R/B by way of the refueling area truck access bay at elevation 3 ft - 7 in. The spent fuel cask is raised from the truck using the spent fuel cask handling crane through the access hatch in the floors at elevation 25 ft - 3 in and 76 ft - 5 in the R/B refueling area.

- The cask is moved to the cask washdown pit and washed to clean off dust and adhered material from the outside surface of the cask.

Table 9.1.2-2 Loads and Load Combinations for New and Spent Fuel ~~Rack~~Storage Racks and Containment Racks DCD_12.03-12.04-40

Load Combination	Acceptance Limit (ASME Section III, Division 1, Article NF3000)
D + L	Level A service limits
D + L + To	
D + L + To +E	
D + L + Ta + E	Level B service limits
D + L + To + Pf	
D + L + Ta + E'	Level D service limits
D + L + Fd	The functional capability of the fuel racks should be demonstrated
Where:	
D :	Dead Loads
L :	Live loads – effect of lifting the empty rack to installation
To :	Thermal effects and loads during normal operating or shutdown conditions, based on the most critical transient or steady state condition.
E :	Loads generated by operating-basis earthquake (OBE)
E' :	Loads generated by SSE
Pf :	Upward force on the racks caused by postulated stuck fuel assembly
Ta	Differential temperature induced loads based on the postulated abnormal design condition (spent fuel rack only)
Fd :	Force caused by the accidental drop of the heaviest load from the maximum possible height

Table 9.1.2-3 Light Load Drop Condition for New and Spent Fuel ~~Rack~~Storage Racks and Containment Racks DCD_12.03-12.04-40

	Drop object	Drop weight	Drop Situation	Drop height above rack top
Case-1N	a fuel assembly plus new fuel handling tool	2,000 lbs	Straight Incline	3.0 feet above rack top
Case-2N			Straight	3.0 feet above rack bottom with empty cell
Case-1S	a fuel assembly plus spent fuel handling tool	2,450 lbs	Straight Incline	2 feet above rack top
Case-2S			Straight	2 feet above rack bottom with empty cell
Case-1C	a fuel assembly plus spent fuel handling tool	2,450 lbs	Straight Incline	3.0 feet above rack top
Case-2C			Straight Incline	3.0 feet above rack bottom with empty cell

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Security-Related Information – Withheld Under 10 CFR 2.390

Figure 9.1.2-3 Location of Containment Racks

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Security-Related Information – Withheld Under 10 CFR 2.390

Figure 9.1.2-4 Arrangement of Containment Racks

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Security-Related Information – Withheld Under 10 CFR 2.390

Figure 9.1.4-2 Section View of Light Load Handling System

shutdown. The sources given are the maximum values with credit for 4 hours of fission and corrosion product decay and purification.

12.2.1.2.2 Reactor Core

Core average gamma ray source strengths are tabulated in Table 12.2-2. These source strengths are used in the evaluation of radiation levels within and around the shutdown reactor.

For source strength calculation, it is assumed that the core has two regions and the irradiation time is 28 months to conservatively bound cycle lengths up to 24 months. The specific power is 32.0 MW/MTU as described in Chapter 4, Table 4.4-1. In this calculation, the specific power was rounded up a fraction to 32.1 MW/MTU. These calculation conditions lead to fission and activation products generated in fuel with burnup of about 55 GWD/MTU in two cycles.

12.2.1.2.3 Spent Fuel

The predominant radioactivity sources in the spent fuel storage and transfer areas in the Reactor Building (R/B) and the refueling cavity and fuel handling areas in the pre-stressed concrete containment vessel (PCCV) are the spent fuel assemblies. The source strengths employed to determine the minimum water depth above spent fuel and shielding walls around the SFP, ~~as well as shielding of~~ the spent fuel transfer tube, refueling cavity, and fuel handling area are tabulated in Table 12.2-54. For the shielding design, the SFP ~~is~~and refueling cavity containment racks are assumed to contain the design maximum number of fuel assemblies. To be conservative, 257 spent fuel assemblies, assumed to be from unloading the full core with only a 24-hour decay period, are assumed to be located in the outer rows of the spent fuel racks. The remaining assemblies, from previous refueling operations, do not significantly affect the shield wall design due to the shielding of the intervening, recently discharged assemblies. To be conservative, six spent fuel assemblies, assumed to be from unloading the core with only a 24-hour decay period, are assumed to fill the containment racks located in the refueling cavity.

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The source strengths in Table 12.2-54 are also used in the evaluation of radiation levels for spent fuel handling, storage, and shipping. These sources are calculated using the ORIGEN code, based on specific power of 32.1 MW/MTU and burnup of 62 GWD/MTU, which is a limitation for maximum burnup for fuel rod as described in Chapter 4, Subsection 4.2.1. Other calculation parameters are tabulated in Table 12.2-70.

12.2.1.2.4 Control Rods, Primary and Secondary Source Rods

As source material, byproduct material or special nuclear material, there are primary and secondary source rods. As described in Chapter 4, Subsection 4.2.2.3 and 4.2.2.3.3, a primary source rod contains californium-252 source, a secondary source rod contains antimony-beryllium source. These rods are stored in the SFP after use. Irradiated control rods are also stored in the SFP. Source strengths of these rods are less than that of spent fuel. Therefore, in radiation shielding design, source strengths of spent fuel are used as these rods' source strengths.

The secondary shield consists of a steel reinforced concrete plate that surrounds the RCS equipment, including piping, pumps, pressurizer, and SGs. This shield protects personnel from the direct gamma ray radiation resulting from reactor coolant activation products and fission products carried away from the core by the reactor coolant. In addition, the secondary shield supplements the primary shield by attenuating neutron and gamma ray radiation escaping from the primary shield. The secondary shield is sized to allow limited access to the containment during full-power operation. The minimum thickness of the secondary shield walls are 4'.

Components of the letdown portion of the CVCS in the containment are located in shielded compartments that are normally over Zone VI, restricted access areas. Shielding is provided for each piece of equipment in the letdown system, consistent with its postulated maximum activity (Section 12.2.1) and with the access and zoning requirements of the adjacent areas. This equipment includes the regenerative heat exchanger, the excess letdown heat exchanger, the letdown heat exchanger, and the letdown lines.

After shutdown, the containment is accessible for limited periods-of-time and all access is controlled. Areas are surveyed to establish allowable working periods. Dose rates are expected to range widely, depending on the location inside the containment (excluding the refueling cavity). These dose rates result from residual fission surveyed from residual fission products, and neutron activation products (components and corrosion products) in the RCS to establish allowable working periods.

Spent fuel is the primary source of radiation during refueling. Because of the high activity of the fission products contained in the spent fuel elements, extensive shielding is provided for areas surrounding the refueling ~~pit~~cavity and the ~~fuel transfer canal~~containment racks. The shielding ensures that radiation levels remain below zone levels specified for the adjacent areas. The water provides the shielding over the spent fuel assemblies during fuel handling. A more detailed description of containment racks used to temporarily store up to six fuel assemblies is provided in Section 9.1. DCD_12.03-12.04-40 DCD_12.03-12.04-42 DCD_12.03-12.04-40 DCD_12.03-12.04-42

12.3.2.2.3 Reactor Building Shielding Design

During normal operations, the major components in the reactor building that contain radioactivity are the RHR, and charging systems. Under accident conditions, the RHR, containment spray and injection systems will contain high levels of radioactivity because these systems will be used following an accident. Charging systems will not be used following an accident, but will be expected to continue to contain same radioactive material as contained during normal operation. Shielding is provided for each piece of equipment consistent with its postulated maximum activity (Section 12.2 of this chapter) and with the access and zoning requirements of the adjacent areas (see Figure 12.3-1).

Depending on the equipment in the compartments, the radiation zones under normal conditions will vary from Zone IV through Zone X. Corridors are generally shielded to allow Zone III access. Operator areas for valve compartments are generally Zone IV for access. Under accident conditions, the radiation levels will be considerably higher (see Section 12.3.1.2.2).

12.3.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

The radiation monitoring system consists of the following:

- Area Radiation Monitoring System (ARMS)
- Airborne Radioactivity Monitoring System
- Process and Effluent Radiation Monitoring System
- Sampling system
- Post-Accident Monitoring Systems (PAM) radiation monitors

The process and effluent radiation monitoring system and sampling systems are described in Chapter 11, Section 11.5.

The PAM are described in Chapter 7, Section 7.5. The portable dose rate and activity monitoring instruments are Type E PAM.

The ARMS and Airborne Radioactivity Monitoring System supplement the personnel and area radiation survey provisions of the plant health physics program described in Section 12.5 and assure compliance with the personnel radiation protection requirements of 10 CFR 20 (Reference 12.3-2), 10 CFR 50 (Reference 12.3-7), 10 CFR 70 (Reference 12.3-21), and the guidelines of RGs 1.21 (Reference 12.3-22), 1.97 (Reference 12.3-23), 8.2 (Reference 12.3-24), and 8.8 (Reference 12.3-1), ANSI N13.1-1999 (Reference 12.3-25), and IEEE 497-2002 (Reference 12.3-28).

The design of the spent fuel ~~pool~~storage racks and containment racks precludes criticality under all postulated normal and accident conditions. Therefore, criticality monitors, as stated in 10 CFR 50.68 (Reference 12.3-26), are not needed. DCD_12.03-12.04-40 DCD_12.03-12.04-42

The ARMS are in conformance with ANSI/ANS HPSSC-6.8.1 (Reference 12.3-27).

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, are to be described by the COL Applicant.

12.3.4.1 Area Radiation Monitoring System

12.3.4.1.1 Design Objectives

The design objectives of the ARMS during normal operating plant conditions and anticipated operational occurrences are as follows:

- To record radiation levels in specific areas of the plant

4.0 DESIGN FEATURES

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent,

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Subsection 9.1.1 of the DCD,

c. A nominal 11.1 inch center to center distance between fuel assemblies placed in spent fuel storage racks.

4.3.1.2 The new fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent,

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Subsection 9.1.1 of the DCD,

c. $k_{eff} \leq 0.98$ if moderated by aqueous foam, which includes an allowance for uncertainties as described in Subsection 9.1.1 of the DCD, and

d. A nominal 16.9 inch center to center distance between fuel assemblies placed in the storage racks.

4.3.1.3 The containment racks are designed and shall be maintained with: DCD_12.03-12.04-40 DCD_12.03-12.04-41

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent,

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Subsection 9.1.1 of the DCD,

c. A nominal 16.9 inch center-to-center distance between fuel assemblies placed in containment racks.

4.3.2 Drainage

Key Activities of POS 4-3:

- The CVCS keeps the RCS inventory above the MCP top level, continuing to POS 4-2. The operation for the removal of RV head stud bolts has been finished.
- The RCS inventory increases to one-foot below the flange level by the RWP pump in order to remove the RV head.
- The operation for the removal of ICIS has been finished, and then the operation for the removal of RV head is started.
- The operation for the hoist of the RV head is started.
- The RV head is transferred for the execution of fuel offload, and the RCS inventory is increased up to cavity full by the CS/RHR pump (End of POS 4-3).

- POS 5: Refueling cavity is filled with water (refueling) – Out of scope of US-APWR LPSD PRA

  POS 5 is period when the refueling cavity is filled with water. To offload fuel from the reactor, the refueling cavity is filled with water. If a loss of decay heat removal were to occur, there is considerable time before the reactor core is exposed due to the boil down of coolant. Therefore, the state in which the refueling cavity is filled with water is identified as one of the states of the plant. The end of POS 5 is defined as the time at which the reactor core is empty.

- POS 6: No fuel in the core or the fuel is partially offloaded – Out of scope of US-APWR LPSD PRA

  POS 6 is the state at which there may be either no fuel in the reactor core or the fuel is partially offloaded. For refueling and examination of fuel, fuel is transported from the RV to the SFP, or temporarily stored in the containment racks during this POS. This state is excluded from the analysis because there is either no fuel in the reactor, or if the fuel is partially offloaded, there is considerable time before the reactor core is exposed given a loss of decay heat removal event. The end of POS 6 is defined as the time at which fuel is loading into the reactor core. DCD_12.03-12.04-40 DCD_12.03-12.04-45

- POS 7: Refueling cavity is filled with water (refueling) – Out of scope of US-APWR LPSD PRA

  POS 7 is the state at which the refueling cavity is filled with water. To load new fuel in the reactor, the refueling cavity is filled with water which defines this POS. If a loss of decay heat removal were to occur, there would be considerable time before the reactor core is exposed by the boiling of coolant. Therefore, the state in which the refueling cavity is filled with water is one of the states of the plant. The end of POS 7 is defined as the time at which the RCS is drained. The change of RCS inventory level is an important factor for LPSD PRA.

- POS 8: RHR cooling (mid-loop operation after refueling)

Table 19.1-81 Disposition of Plant Operating States for LPSD PRA (Sheet 1 of 2)

POS	Description	POS modeled	Reason for model exclusion
1	Low power operation	No	This POS is a low power shutdown state and ECCS actuation signal is available. Further, all components will not be planned to be maintenance in this POS. Therefore, the risk of this POS is included in full power PRA
2	Hot standby condition	No	This POS is a hot standby state before RHR cooling and ECCS actuation signal is available. Further, all components will not be planned to be maintenance in this POS. Therefore, the risk of this POS is included in full power PRA.
3	RHR cooling (RCS full)	Yes	N/A
4	RHR cooling (mid-loop operation)	Yes	N/A
5	Refueling cavity is filled with water (refueling)	No	This POS is the state that refueling cavity is filled with water. Since there is large inventory water in the cavity, there would be sufficient time by core exposure and operator action will be more reliable. CDF during this POS is considered negligible.
6	No fuels in the core, or the core is partially offloaded.	No	This POS is the state at which there is either no fuel in the reactor core or the fuel is partially offloaded. For refueling and examination of the fuel, the fuel is transported from the RV to the spent fuel pit, or temporarily stored in the containment racks during this POS. This state is excluded from the analysis because there is either no fuel in the reactor, or if the fuel is partially offloaded, there is considerable time before the reactor core is exposed given a loss of decay heat removal event. The end of this POS is defined as the time at which fuel is fully loaded into the reactor core.
7	Refueling cavity is filled with water (refueling)	No	This POS is the state that refueling cavity is filled with water. Since there is large inventory in the cavity, there would be sufficient time by core exposure and operator action will be more reliable. CDF during this POS is considered negligible.
8	RHR cooling (mid-loop operation)	Yes	N/A
9	RHR cooling (RCS full)	Yes	N/A

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