RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

11/22/2013

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 1055-7184 REVISION 0**
SRP SECTION:	**09.01.02 - New and Spent Fuel Storage**
APPLICATION SECTION:	**1.9,3.8,5.1,9.1,12.3**
DATE OF RAI ISSUE:	**10/07/2013**

QUESTION NO.: 09.01.02-49:

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 906-6332 Question 09.01.02-26, dated May 23 2013, included Technical Report MUAP-13012-P (R0) “Mechanical Analysis for US-APWR Containment Racks” MUAP-13012-P (R0) Section 4.1(2) “Straight Deep Drop Event” states that deep drop scenarios do not need to be analyzed for the containment racks since the containment racks do not have an elevated baseplate or rack pedestals. However, unlike the spent fuel pool, the dummy fuel element may be moved and lifted without being surrounded by water. Since MUAP-13012-P (R0) is silent with respect to dry handling and storage of the dummy bundle in the refueling cavity, it is not clear to the staff that the analysis bounds scenarios that have the potential to cause liner damage.
Please revise and update US-APWR MUAP-13012-P to utilize calculation assumptions that are consistent with relevant industry operating experience related to potential accident conditions, or provide the specific alternative approaches used and the associated justification.

ANSWER:

The light load handling system includes the new fuel elevator, fuel handling machine, refueling machine, the suspension hoist of the spent fuel cask handling crane, fuel transfer system, and various fuel handling tools. The refueling machine transports fuel assemblies between the fuel transfer system and the reactor core within the confines of the refueling cavity.

In DCD Revision 4, subsection 9.1.4.1 “Design Bases” the fourth bullet in the first paragraph describes design requirements in accordance with ANSI/ANS57.1-1992 which are applicable to the LLHS design and says that “- The LLHS components involved in grappling, latching, translating, rotating, supporting, or hoisting fuel assemblies are designed to assure there will not be a structural failure of any part of the handling equipment, which would result in dropping or damaging a fuel assembly.”

In addition to that, in DCD Revision 4, subsection 9.1.4.3 “Safety Evaluation” fifth bullet in the first paragraph states, “Fuel handling performance is assured to be within acceptable limits by designing and configuring the light load handling system to comply with ANS 57.1-1992 (Ref. 9.1.7-13).”.

The light load handling system is evaluated as to its ability to assure there is no risk of dropping a transporting component, including a dummy fuel element. Therefore, there is no potential to cause liner damage.

Impact on DCD

There is no impact on the DCD.

Impact on R-COLA

There is no impact on the R-COLA.

Impact on PRA

There is no impact on the PRA.

Impact on Topical Report / Technical Report

There is no impact on the Topical Report / Technical Report because the discussion related to the dropping of a load which causes liner damage is not in the scope of the MUAP-13012-P and the discussion in DCD 9.1.4 as described in this RAI response provides the basis.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

11/22/2013

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 1055-7184**
SRP SECTION:	**09.01.02 - New and Spent Fuel Storage**
APPLICATION SECTION:	**1.9, 3.8, 5.1, 9.1, 12.3**
DATE OF RAI ISSUE:	**10/07/2013**

QUESTION NO. : 09.01.02-50

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. 10 CFR 50.68(b)(4) requires that if no credit for soluble boron is taken, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, if flooded with unborated water. SRP 9.1.1 “Criticality Safety of Fresh and Spent Fuel Storage and Handling,” adds clarification by stating that when fully loaded and flooded with full-density unborated water, the K(eff) will not exceed 0.95 for all normal and credible abnormal conditions.

In RAI 895-6172 Question 12.03-12.04-41, dated 27 January 2012, the staff asked the applicant to describe how the containment fuel racks comply with the requirements of 10 CFR 50.68(b)(4). The applicant’s response to RAI 895-6172 Revision 3 Question 12.03-12.03-41, dated 25 April 2012, stated that 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, Technical Specification (TS) 3.9.1 needs no modification. TS 3.9.1 is only applicable during MODE 6. MODE 6 is not applicable when all fuel is out of the reactor vessel. However, the applicant’s response to RAI 906-6332 Question 09.01.02-26, dated May 23 2013, included Technical Report MUAP-13011-P (R0) “Criticality Analysis for US-APWR Containment Racks.” MUAP-13011-P (R0) Subsection 2.3.1.3.2 “Mislocated Fresh Fuel Assembly,” states that “The mislocation of a fresh fuel assembly could, in the absence of soluble neutron absorber, result in exceeding the regulatory limit (keff < 0.95). In addition, MUAP-13011-P (R0) Table 2-7 “Summary of CR Accident Case Calculations,” shows that the Maximum keff (0 ppm soluble boron) is 1.0452.

Please revise and update the US-APWR DCD Technical Specifications section 3.9.1 to reflect the requirement for maintaining soluble boron reactivity controls consistent with 10 CFR 50.68(b)(4), when fuel is in the containment racks, and no fuel is in the reactor vessel, or provide the specific alternative approaches used and the associated justification.

ANSWER:

Under accident conditions, keff shall be maintained less than 0.95 (0.9269) with 800 ppm soluble boron as shown in Table 2-7 of Technical Report MUAP-13011-P (R0) "Criticality Analysis for US-APWR Containment Racks." In addition, under normal conditions, keff is maintained less than 0.95 (0.8991) with unborated water as shown in Table 2-6 of Technical Report MUAP-13011-P (R0). The containment rack design is adequate to meet the requirement of 10 CFR 50.68(b)(4).

As described in the response to Question 09.01.02-53 of this RAI, DCD Rev.4 Chapter 16 TS 3.9.1, Boron Concentration during Refueling, will be revised to address the condition when one or more fuel assemblies are seated in the Containment Racks. This will ensure that the boron concentration is maintained adequate to be consistent with the criticality analysis and results in MUAP-13011. In addition, MUAP-13011 will be revised to indicate that the boron concentration is controlled by Technical Specifications.

Impact on DCD

Refer to the response to Question 09.01.02-53.

Impact on R-COLA

There is no impact on the R-COLA.

Impact on PRA

There is no impact on the PRA.

Impact on Technical / Topical Report

Technical Report MUAP-13011 will be revised as shown in the attached markup.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

11/22/2013

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 1055-7184**
SRP SECTION:	**09.01.02 – New and Spent Fuel Storage**
APPLICATION SECTION:	**1.9, 3.8, 5.1, 9.1, 12.3**
DATE OF RAI ISSUE:	**10/07/2013**

QUESTION NO. 09.01.02-51

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. 10 CFR 50.68(b)(4) requires that if no credit for soluble boron is taken, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, if flooded with unborated water. SRP 9.1.1 “Criticality Safety of Fresh and Spent Fuel Storage and Handling,” adds clarification by stating that when fully loaded and flooded with full-density unborated water, the K(eff) will not exceed 0.95 for all normal and credible abnormal conditions.

In RAI 895-6172 Question 12.03-12.04-40 dated 27 January 2012 the staff asked the applicant to describe how the containment fuel racks comply with the requirements of 10 CFR 50.68(b)(4). The applicant’s response to RAI 895-6172 Revision 3 Question 12.03-12.03-40, dated 25 April 2012, committed to adding Technical Specifications subsection 4.3.1.3, including the statement that the containment racks are designed and shall be maintained with keff less than 0.95 if fully flooded with unborated water. However, the applicant’s response to RAI 906-6332 Question 09.01.02-26, dated May 23 2013, included Technical Report MUAP-13011-P (R0) “Criticality Analysis for US-APWR Containment Racks.” MUAP-13011-P (R0) Subsection 2.3.1.3.2 “Mislocated Fresh Fuel Assembly,” states that “The mislocation of a fresh fuel assembly could, in the absence of soluble neutron absorber, result in exceeding the regulatory limit (keff < 0.95). In addition, MUAP-13011-P (R0) Table 2-7 “Summary of CR Accident Case Calculations,” shows that the Maximum keff (0 ppm soluble boron) is 1.0452.

Please revise and update the US-APWR DCD Technical Specifications to reflect the requirement for maintaining reactivity controls consistent with 10 CFR 50.68(b)(4), or provide the specific alternative approaches used and the associated justification.

ANSWER:

Under accident conditions, keff shall be maintained less than 0.95 (0.9269) with 800 ppm soluble boron as shown in Table 2-7 of Technical Report MUAP-13011-P (R0) “Criticality Analysis for US-APWR Containment Racks.” The boron concentration of the refueling cavity water is controlled by Technical Specification (TS) 3.9.1 which will be revised as described in the response to Question 09.01.02-53 of this RAI. As described in response to Question 09.01.02-53, the applicability statement is revised to address the condition when the Containment Racks are in use so that it is clear that this requirement applies under conditions like those analyzed in MUAP-13011. In addition, during normal conditions, keff shall be maintained less than 0.95 (0.8991) with unborated water as shown in Table 2-6 of Technical Report MUAP-13011-P (R0). The Containment Rack design is adequate to meet the requirement of 10 CFR 50.68(b)(4), and TS 4.3.1.3 and Technical Report MUAP-13011 will be revised to reflect this response.

Impact on DCD

DCD Chapter 16 TS 4.3.1.3 will be revised as shown in the attached markup. Other changes in the TS refer to the response to Question 09.01.02-53.

Impact on R-COLA

There is no impact on the R-COLA.

Impact on PRA

There is no impact on the PRA.

Impact on Technical/Topical Report

Refer to the response to Question 09.01.02-50.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

11/22/2013

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 1055-7184**
SRP SECTION:	**09.01.02 – New and Spent Fuel Storage**
APPLICATION SECTION:	**1.9, 3.8, 5.1, 9.1, 12.3**
DATE OF RAI ISSUE:	**10/07/2013**

QUESTION NO. 09.01.02-52

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. 10 CFR 50.68(b)(4) requires that if no credit for soluble boron is taken, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, if flooded with unborated water. SRP 9.1.1 “Criticality Safety of Fresh and Spent Fuel Storage and Handling,” adds clarification by stating that when fully loaded and flooded with full-density unborated water, the K(eff) will not exceed 0.95 for all normal and credible abnormal conditions.

In RAI 895-6172 Question 12.03-12.04-40 dated 27 January 2012 the staff asked the applicant to describe how the containment fuel racks comply with the requirements of 10 CFR 50.68(b)(4). The applicant’s response to RAI 895-6172 Revision 3 Question 12.03-12.03-40, dated 25 April 2012, committed to changing subsection DCD 3.1.6.3.1 “Discussion,” to stated that the containment racks are designed to have sufficient separation between adjacent fuel assemblies so the maximum keff 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. However, the applicant’s response to RAI 906-6332 Question 09.01.02-26, dated May 23 2013, included Technical Report MUAP-13011-P (R0) “Criticality Analysis for US-APWR Containment Racks.” MUAP-13011-P (R0) Subsection 2.3.1.3.2 “Mislocated Fresh Fuel Assembly,” states that “The mislocation of a fresh fuel assembly could, in the absence of soluble neutron absorber, result in exceeding the regulatory limit (keff < 0.95). In addition, MUAP-13011-P (R0) Table 2-7 “Summary of CR Accident Case Calculations,” shows that the Maximum keff (0 ppm soluble boron) is 1.0452.

Please revise and update the US-APWR DCD to reflect the requirement for maintaining reactivity controls consistent with 10 CFR 50.68(b)(4), or provide the specific alternative approaches used and the associated justification.

ANSWER:

Under accident conditions, keff shall be maintained less than 0.95 (0.9269) with 800 ppm soluble boron as shown in Table 2-7 of Technical Report MUAP-13011-P (R0) “Criticality Analysis for US-APWR Containment Racks.” The boron concentration of the refueling cavity water is controlled by Technical Specification (TS) 3.9.1 which will be revised as described in the response to Question 09.01.02-53 of this RAI. As described in response to Question 09.01.02-53, the applicability statement is revised to address the condition when the Containment Racks are in use so that it is clear that this requirement applies under conditions like those analyzed in MUAP-13011. In addition, during normal conditions, keff shall be maintained less than 0.95 (0.8991) with unborated water as shown in Table 2-6 of Technical Report MUAP-13011-P (R0). The Containment Rack design is adequate to meet the requirement of 10 CFR 50.68(b)(4), and DCD 3.1.6.3.1 and Technical Report MUAP-13011 will be revised to reflect this response.

Impact on DCD

DCD Subsection 3.1.6.3.1 will be revised as shown in the attached markup. Other changes in the TS refer to the response to Question 09.01.02-53.

Impact on R-COLA

There is no impact on the R-COLA.

Impact on PRA

There is no impact on the PRA.

Impact on Technical/Topical Report

Refer to the response to Question 09.01.02-50.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

11/22/2013

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 1055-7184**
SRP SECTION:	**09.01.02 - New and Spent Fuel Storage**
APPLICATION SECTION:	**1.9, 3.8, 5.1, 9.1, 12.3**
DATE OF RAI ISSUE:	**10/07/2013**

QUESTION NO. : 09.01.02-53

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. 10 CFR 50.68(b)(4) requires that if no credit for soluble boron is taken, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, if flooded with unborated water. SRP 9.1.1 “Criticality Safety of Fresh and Spent Fuel Storage and Handling,” adds clarification by stating that when fully loaded and flooded with full-density unborated water, the K(eff) will not exceed 0.95 for all normal and credible abnormal conditions.

In RAI 895-6172 Question 12.03-12.04-41 dated 27 January 2012 the staff asked the applicant to describe how the US-APWR DCD Technical Specification ensured compliance with the requirements of 10 CFR 50.68(b)(4) when fuel is in the containment racks. The applicant’s response to RAI 895-6172 Revision 3 Question 12.03-12.03-41, dated 25 April 2012, contained statements such as “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,” and “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.” US-APWR DCD Revision 3 subsection 1.1. “Definitions,” MODE which 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 with fuel in the reactor vessel. The response to RAI 895-6172 Revision 3 Question 12.03-12.03-41 did not commit to changing the definition of MODE 6, therefore, the noted response to RAI 895-6172 Revision 3 Question 12.03-12.03-41 is not consistent with the definition of MODE 6 when fuel is out of the reactor vessel, but present in the containment racks.

Please revise and update the US-APWR DCD Technical Specifications definition of MODE 6 or add a TS Applicability statement in the applicable TS (e.g. TS 3.7.10, 3.8.2, 3.8.5, 3.8.8, 3.8.10, 3.9.1, 3.9.2, 3.9.3, 3.9.4, and 3.9.7) to address “when one or more irradiated or new fuel assemblies are seated in the refueling cavity containment racks,” that reflect the requirements for controls and storage of fuel consistent with 10 CFR 50.68(b), or provide the specific alternative approaches used and the associated justification.

ANSWER:

MHI agrees that the current Mode 6 definition does not address the condition when no fuel is in the reactor vessel and fuel is in the containment racks. Notwithstanding, MHI will not change the definition of Mode 6. Instead, MHI will revise the applicability of specific Technical Specifications (TS) to indicate that they are also applicable when fuel is seated in the containment racks. This will also ensure that the presence of fuel in the containment racks is treated in a manner similar to the fuel in the SFP.

The TS that MHI will revise are: 3.8.2, 3.8.5, 3.8.8, 3.8.10, 3.9.1, 3.9.2, 3.9.5, and 3.9.7. The corresponding BASES of these TS will also be revised. In addition, Table 3.3.2-1 of TS 3.3.2 and the corresponding BASES will also be revised. Lastly, the description in the BASES of LCO 3.0.3 will be revised for consistency.

These TS changes are consistent with the requested changes in this question with the exception of TS 3.7.10, 3.9.3, and 3.9.4. These three TS apply only during movement of irradiated fuel assemblies to address the possibility of a fuel handling accident (FHA). When fuel is seated in the containment racks and <u>not</u> being moved, these three TS are not required. If the operators attempted to move fuel into or out of the containment racks, then these TS would apply. Therefore, MHI did not modify the applicability of TS 3.7.10, 3.9.3, or 3.9.4.

Impact on DCD

DCD Chapter 16 Technical Specifications will be revised as shown in the attached markup.

Impact on R-COLA

The R-COLA Technical Specifications will require revision to be consistent with the DCD.

Impact on PRA

There is no impact on the PRA.

Impact on Technical / Topical Report

There is no impact on any Technical / Topical Report.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

11/22/2013

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 1055-7184 REVISION 0**
SRP SECTION:	**09.01.02- NEW AND SPENT FUEL STORAGE**
APPLICATION SECTION:	**9.1**
DATE OF RAI ISSUE:	**10/07/2013**

QUESTION NO.: 09.01.02-56

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) 63 "Monitoring fuel and waste storage," requires that the fuel storage system be designed for adequate safety under anticipated operating and accident conditions. GDC 64 "Monitoring radioactivity releases," requires systems to monitor for excessive radiation levels in the reactor containment atmosphere and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents, and to initiate appropriate safety actions.

US-APWR DCD Revision 3, Section 3.1.6.4 Criterion 63 – “Monitoring Fuel and Waste Storage,” discusses area radiation monitoring and airborne radioactivity monitoring for fuel stored in the Spent Fuel Pool and new fuel storage racks, however, this section does not discuss how the US-APWR design addressed radiation monitoring for GDC 63 and GDC 64 for fuel stored in the containment racks.

Please revise and update the US-APWR DCD section 3.1.6.4 to describe the radiation monitoring instruments provided to meet GDC 63 and GDC 64 for fuel located in the containment racks, or provide the specific alternative approaches used and the associated justification.

ANSWER:

In response to RAI No. 1055-7184, Question No. 09.01.02-47, DCD Subsection 12.3.4.1.2 was revised to indicate that portable area radiations monitors (ARMs) would be utilized to warn operators or personnel of deteriorating conditions when working on the refueling platform and refueling cavity when fuel is stored in the containment racks.

DCD Subsections 9.1.4.2.1.13 and 9.1.4.2.2.2 describe how the refueling cavity water level instruments and alarms shown on Figure 5.1-2 (Sheet 3 of 3) warn personnel of deteriorating conditions, thus keeping radiation dose ALARA in the fuel handling area and refueling cavity. DCD Subsection 11.5.3 describes containment radiation monitors required by GDC 64.

DCD Subsections 3.1.6.4 and 3.1.6.5 have been revised to describe the portable ARM used to warn personnel of deteriorating conditions in the refueling cavity near the containment racks when fuel is temporarily stored, and the refueling cavity water level instrumentation that annunciates in the MCR to demonstrate compliance with GDCs 63 and 64.

Impact on DCD

DCD Subsections 3.1.6.4 and 3.1.6.5 will be revised as shown in the attached markup.

Impact on R-COLA

There is no impact on the R-COLA.

Impact on PRA

There is no impact on the PRA.

Impact on Technical/Topical Report

There is no impact on Technical/Topical Report.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

11/22/2013

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 1055-7184**
SRP SECTION:	**09.01.02 – New and Spent Fuel Storage**
APPLICATION SECTION:	**1.9, 3.8, 5.1, 9.1, 12.3**
DATE OF RAI ISSUE:	**10/07/2013**

QUESTION NO. 09.01.02-59

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. 10 CFR 50.68(b)(4) requires that if no credit for soluble boron is taken, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, if flooded with unborated water. SRP 9.1.1 “Criticality Safety of Fresh and Spent Fuel Storage and Handling,” adds clarification by stating that when fully loaded and flooded with full-density unborated water, the K(eff) will not exceed 0.95 for all normal and credible abnormal conditions.

In RAI 895-6172 Question 12.03-12.04-40 dated 27 January 2012 the staff asked the applicant to describe how the containment fuel racks comply with the requirements of 10 CFR 50.68(b)(4). The applicant’s response to RAI 895-6172 Revision 3 Question 12.03-12.03-40, dated 25 April 2012, committed to issuing three new Technical Reports describing the Containment Rack design basis. The applicant’s response to RAI 906-6332 Question 09.01.02-26, dated May 23 2013, included Technical Report MUAP-13011-P (R0) “Criticality Analysis for US-APWR Containment Racks.” However, MUAP-13011-P (R0) subsection 2.3.1.3 “Abnormal Location of a Fuel Assembly,” subsection 2.3.1 “Abnormal and Accident Conditions,” and MUAP-13011-P (R0) Figure 2-4 “MCNP Model for Fuel Displacement with Cells or CR,” do not provide a description of the limiting locations of fuel located in the racks and the dropped bundle.

Please revise and update Technical Report MUAP-13011-P to describe the limiting arrangement of fuel used for performing the analysis for ensuring reactivity controls are maintained consistent with 10 CFR 50.68(b)(4) when analyzing for a dropped fuel bundle with fuel in the containment racks, or provide the specific alternative approaches used and the associated justification.

ANSWER:

The analysis result of the limiting case of abnormal fuel location in the racks is summarized in Table 2-7 of Technical Report MUAP-13011-P (R0) "Criticality Analysis for US-APWR Containment Racks." The corresponding analysis model is shown in Figure 09.01.02-59.1. This case assumes mislocation of a fuel assembly. The fuel assembly is never mislocated between cells, since the separation between cells is narrower than the width of an assembly. The case of a drop accident does not result in an increase in reactivity, and therefore it is not considered.

Technical Report MUAP-13011 will be revised to include this figure of the analysis model.

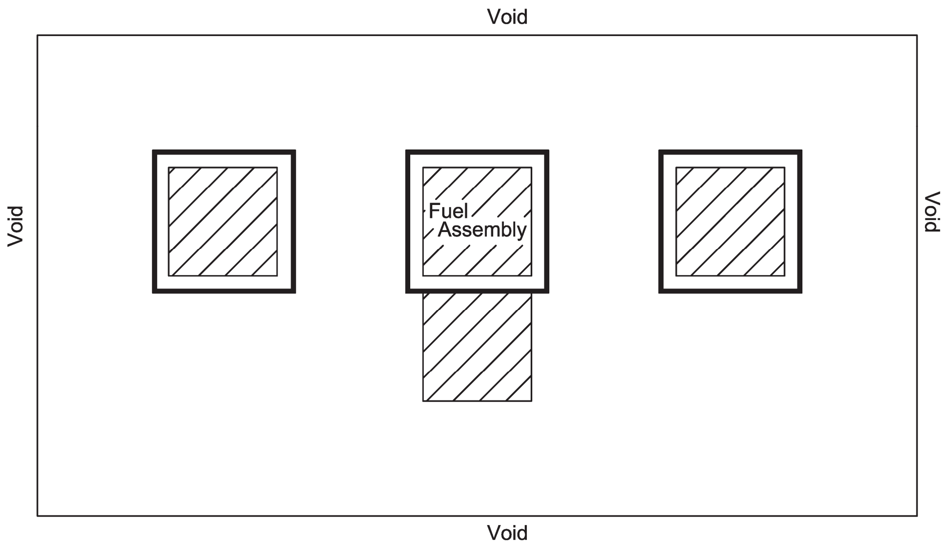


Figure 09.01.02-59.1 Analysis model for abnormal fuel location in the racks

Impact on DCD

There is no impact on the DCD.

Impact on R-COLA

There is no impact on the R-COLA.

Impact on PRA

There is no impact on the PRA.

Impact on Technical/Topical Report

Technical Report MUAP-13011 will be revised as shown in the attached markup.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

11/22/2013

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 1055-7184**
SRP SECTION:	**09.01.02 - New and Spent Fuel Storage**
APPLICATION SECTION:	**1.9, 3.8, 5.1, 9.1, 12.3**
DATE OF RAI ISSUE:	**10/07/2013**

QUESTION NO. : 09.01.02-60

The applicant's response to RAI 895-6172, Question 12.03-12.04-41, dated April 24, 2012, provided additional information regarding Technical Specifications (TS) for the use of Containment Racks (CR) as temporary storage locations. Specifically, the applicant stated TS 3.9.5, Residual Heat Removal (RHR), is not necessary in order to cool the six assemblies located in the CR because the buildup of decay heat is not sufficient to require RHR cooling. However, the applicant has not provided an adequate heat up analysis that demonstrates the assemblies located in the refueling cavity, and the bulk refueling cavity temperature, will remain below acceptable temperatures without RHR cooling.

Compliance with GDC 61 requires, in part, that fuel storage be designed to ensure adequate safety under normal and postulated accident conditions, this includes suitable shielding for radiation protection, residual heat removal, and the capability to prevent a significant reduction in fuel storage coolant inventory under accident conditions.

The staff requests the applicant to provide an analysis that demonstrates all fuel assemblies located in the refueling cavity will remain protected under all operating conditions. RG 1.13, "Spent Fuel Storage Facility Design Basis," provides guidance for protection of stored fuel; this includes temporarily stored fuel in the CR. The analysis should be consistent with the methodologies of RG 1.13 and consider the following:

a. The maximum expected number of potential assemblies (e.g. six in the CR, and one in transfer or in the upender).

b. Since a seismic category I system (RHR) is not credited for cooling the refueling cavity for all operational conditions, demonstrate that the bulk refueling cavity temperature remains below acceptable limits under normal and abnormal conditions (justify limits). The refueling cavity should have a seismic category I makeup water system and backup makeup system with sufficient makeup capacity higher than the worse boiloff rate. Specify the seismic category I water source credited for maintaining cavity water level above TS 3.9.7 limits. Also, spent fuel pool pumps should not be credited unless there is a TS requirement to have the fuel transfer tube valve/gate open when there are fuel assemblies in the refueling cavity.

c. The analysis should calculate the time to exceed bulk refueling cavity design temperature without RHR cooling in both normal operation (i.e. cavity water level at TS minimum), and accident conditions (e.g. failure of a non-seismic component, such as a nozzle dam, resulting in refueling cavity water level to drain to the reactor vessel flange) to ensure plant personnel have adequate time to respond to the event. Justification is necessary for an assumed starting water temperature above 140^0F.

d. Discuss how the containment building and its ventilation and filtration system will handle the boiling water of the refueling cavity when in no mode (i.e. no fuel in the vessel, and up to seven assemblies in the refueling cavity).

In addition, DCD Revision 3, Section 5.4.7.2.3.5 states, “During refueling, the RHRS is maintained in-service with the number of pumps and heat exchangers in operations as required by the heat load.” This statement is vague and it is unclear to the staff what heat loads would require RHRS to be in-service. This statement and any other affected DCD sections should be revised or updated based on the analysis discussed above regarding the protection of fuel assemblies located in the refueling cavity.

ANSWER:

As described in the response to Question 09.01.02-53 of this RAI, MHI will revise TS 3.9.5 to clarify that the TS does apply when irradiated fuel is in the containment racks.

Despite this change to the TS, the temperature analysis is provided below to show that the refueling cavity water temperature will remain below a reasonable value (200 ^{0}F) under all operating conditions during a loss of RHR cooling.

The heat up analysis of the time from the initial temperature to reach the refueling cavity temperature of 200 ^{0}F, which is specified by RG 1.13 (that is the same as the refueling cavity design temperature), is provided below. The analysis assumes the following conditions that bound all modes.

Input Conditions

The following are the analysis conditions used to conservatively evaluate the time for the refueling cavity temperature to reach 200 ^{0}F.

- The maximum bulk water temperature in the refueling cavity (i.e. initial water temperature for this analysis) is assumed to be 140 ^{0}F which is based on the requirement of limiting temperature during refueling specified by RG 1.13.

- The refueling cavity water capacity is assumed to be approximately 1440 m^3 for this analysis. This capacity is based on when the water level is at 23 ft above the RV flange level, which is a level administratively controlled by TS 3.9.7. Note that the response to Question 09.01.02-53 of this RAI provides changes to TS 3.9.7 to clarify that it is applicable when irradiated fuel is in the containment racks. Although the refueling cavity is connected to the refueling canal through the fuel transfer tube during refueling operation, this evaluation conservatively assumes that the refueling cavity is physically separated from the refueling canal. For the reasons described in the response to Question 09.01.02-46 of this RAI, the drain down event from the steam generator (SG) manways is not considered in this analysis.

- Seven irradiated fuel assemblies are assumed to be located in the refueling cavity. As described in DCD Rev.4 Subsection 9.1.2.2.3, the US-APWR has two containment racks in the refueling cavity that can temporarily store a total of six irradiated fuel assemblies. Additionally, one fuel assembly in transfer or in the upender is assumed. Thus, seven fuel assemblies (the maximum number) are assumed.

- The decay heat at 108 hours after shutdown, which is the time of initiation of the fuel offload as described in DCD Rev.4 Table 19.1-82, is assumed for this analysis. Since twice burned fuel assemblies generate more decay heat than once burned fuel assemblies, the decay heat from the assemblies in the containment racks (and in transfer or upender) is calculated by conservatively assuming the 7 assemblies are twice burned assemblies. At the end of a typical fuel cycle, approximately half of the total number of fuel assemblies will be twice burned. Since the total number of fuel assemblies in the core is 257 from DCD Rev.4 Table 4.1-1, that means that there are approximately 128.5 twice burned fuel assemblies. The decay heat from these 128.5 twice burned assemblies at 108 hours after shutdown is approximately 3.00×10^7 BTU/h (8.8 MW). Therefore, the decay heat from fuel assemblies in the containment racks (and in transfer or upender) is less than 0.5 MW (=8.8 MW x 7 / 128.5). Although the calculated value is less than 0.5 MW, this analysis conservatively assumes 1.0 MW as the decay heat.

- This analysis does not assume attenuation of the decay heat.

- Heat release from the water surface of the refueling cavity and heat transfer to the structures of the refueling cavity (i.e. concrete, etc.) are conservatively not taken into account.

Analysis, Result and Conclusion

The time to increase from the initial temperature (140 ^{0}F) to 200 ^{0}F is calculated as below:

$$t_1 = V_{refueling\ cavity} \times \rho \times (h_{200} - h_{140}) / (Q_D \times 1000 \times 3600)$$

where,

t_1 : Time for temperature rise (hour)
$V_{refueling\ cavity}$: Capacity of the refueling cavity (m^3)
ρ : Density (kg/m^3)
h_{200} : Enthalpy at 200 ^{0}F (kJ/kg)
h_{140} : Enthalpy at 140 ^{0}F (kJ/kg)
Q_D : Decay heat from seven fuel assemblies (MW)
Note that 1000 and 3600 are unit conversion factors

The time for the temperature to increase from the initial water temperature to 200 ^{0}F is calculated to be approximately 53 hours. The duration of 53 hours is more than adequate to allow operators to take effective appropriate action to open the fuel transfer gate valve and the SFP weir gate between the refueling canal and SFP in case that the gate valve and the weir gate are closed. When the gate valve and the weir gate are opened, the makeup water from the EFW pit, which is one of the seismic category I water sources for the SFP, is available to be supplied to the refueling cavity through the SFP and the refueling canal.

Therefore, the design of the refueling cavity satisfies the requirement of RG 1.13 and an additional makeup line from a seismic category I water source is not necessary for the refueling cavity. Thus the integrity of fuel assemblies temporarily stored in the containment racks is maintained during a loss of RHR cooling.

As described above, there is sufficient time for taking the appropriate action to prevent reaching the design temperature of the refueling cavity. However, by any remote chance, in the event of the boiling of the water in the refueling cavity during fuel handling operation, Technical Specification 3.9.5 requires the containment equipment hatch and containment air locks to be closed to prevent the boiling water from being released outside of the containment. The purge lines of the ventilation and filtration system shall also be closed during this event.

The RHRS will be controlled by TS during refueling, including when irradiated fuel is in the containment racks. This is consistent with the statement in DCD Section 5.4.7.2.3.5 which states that the RHRS is maintained in service. Therefore, there is no need to revise the current description in the DCD.

Impact on DCD

Refer to the response to Question 09.01.02-53.

Impact on R-COLA

There is no impact on the R-COLA.

Impact on PRA

There is no impact on the PRA.

Impact on Technical / Topical Report

There is no impact on any Technical / Topical Report.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

11/22/2013

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 1055-7184**
SRP SECTION:	**09.01.02 - New and Spent Fuel Storage**
APPLICATION SECTION:	**1.9, 3.8, 5.1, 9.1, 12.3**
DATE OF RAI ISSUE:	**10/07/2013**

QUESTION NO. : 09.01.02-61

Follow-up to Thermal-Hydraulic Analysis for US-APWR Containment Racks

On May 21, 2013, the staff received a technical report titled, “Thermal-Hydraulic Analysis for US-APWR Containment Racks.” This report concluded no local boiling occurs anywhere along the fuel rods in the Containment Racks (CR), and the peak fuel cladding temperatures for all of the stored assemblies remains below the local saturation temperature of the surrounding water. However, the staff cannot make the same conclusion because some of the assumptions in the analysis are missing or are not adequately justified.

The staff requests the applicant to provide additional information regarding the following items:

a. On page 5, the analysis states that the radial peaking factor is taken into consideration as part of the maximum decay heat per fuel rod (Qrod), however, this value is missing from Table 6-1. What is the maximum decay heat per fuel rod without radial or axial peaking factors, and what are the assumptions for that value (i.e. how many hours after shutdown is assumed before the assembly is moved to the CR).

b. The analysis uses an initial bulk water temperature of 160^0F. However, this temperature is inconsistent with RG 1.13, “Spent Fuel Storage Facility Design Basis,” which states bulk water temperature for stored fuel should remain below 140^0F, and DCD Revision 3, Section 3.8.1.5.3, which states for normal operation concrete temperatures are not to exceed 150^0F. It is unclear to the staff if the use of 160^0F in the analysis is for conservatism or is an actual expected operating temperature. If the 160^0F is only for conservatism, what is the maximum normal operating bulk water temperature of the refueling cavity?

c. On page 11, the analysis states the minimum depth of water at the top of active fuel length is approximately 29 feet. It is not clear to the staff whether this depth is calculated from normal water level or the minimum required water level by TS 3.9.7 on Refueling Cavity Water Level (23 feet above the vessel flange). Justify the use of a minimum water depth of 29 feet and why it is conservative.

ANSWER:

Additional information and justification for some inputs described in the technical report entitled “Thermal-Hydraulic Analysis for US-APWR Containment Racks, MUAP-13013 R0” are provided below:

a. The radial peaking factor is 1.78 which has been described in the Table 6-1 of the technical report MUAP-13013 R0 as “Radial Power Peaking Factor”. The maximum decay heat per fuel rod is approximately 2.21×10^5 BTU/h. The decay heat is based on the fuel at 126 hours after shutdown (i.e. at the time of the maximum bulk water temperature) and the maximum fuel assembly burn-up is 62,000 MWd/MTU. Other parameters are shown in Table 6-1 of the technical report “Thermal-Hydraulic Analysis for US-APWR Spent Fuel Racks MUAP-09014 R0”. Parameters in the table for the spent fuel racks are the same input conditions as the decay heat for the analysis for the containment racks.

   As stated above, the analysis used the decay heat at the time when the local water temperature becomes the maximum (126 hours) and not that of the time immediately after the start of the refueling operation. This is done for conservatism in terms of the local water temperature. The local water temperature becomes the maximum when the initial water temperature, that is, the bulk water temperature of the refueling cavity, becomes the maximum. Also, in the analysis, the bulk water temperature in the refueling cavity is conservatively assumed to be the same as that in the spent fuel pit, although the bulk water temperature in the spent fuel pit is higher in reality. The bulk water temperature of the spent fuel pit peaks at 126 hours after reactor shutdown as shown in Table 7-2 of the technical report MUAP-09014 R0. Therefore, the use of decay heat at 126 hours after reactor shutdown is appropriate for this analysis.

b. The temperature of 160^0F is set conservatively for the analysis, and it is not an actual expected operating temperature of the refueling cavity. The temperature of 160°F that is used in the analysis for containment racks is the value that is used in the analysis for the spent fuel racks as well. This 160^0F is obtained from the SFP bulk temperature at the time of accident, and is adequately conservative for the analysis of containment racks. The maximum bulk water temperature of the refueling cavity during normal operation is considered to be equivalent to that of the SFP since it is connected to the SFP, and this temperature is approximately 126^0F. Note that this temperature is less than the values discussed in RG 1.13 and DCD Section 3.8.1.5.3.

c. The minimum depth of approximately 29 ft for the analysis is based on the height between the low water level setpoint and the top of the containment racks. The refueling cavity water level is controlled by TS 3.9.7. Note that the response to Question 09.01.02-53 of this RAI provides changes to TS 3.9.7 to clarify that it is applicable when irradiated fuel is in the containment racks. The water level is monitored by the refueling cavity water level instrument RCS-LIA-011-N and this instrument has a low water level alarm. So in the event that the refueling cavity water level decreases, the operator can take action to recover the refueling cavity water level. In addition, when the fuel transfer gate valve and the SFP weir gate between the refueling canal and SFP are open, the narrow range SFP water level instruments (SFS-LICA-010-S and SFS-LICA-020-S) are also available to detect a water level decrease of the refueling cavity. Therefore, since there is a sufficient capability to detect such abnormal condition, the assumption of the minimum depth of 29 ft is appropriate for this analysis.

Impact on DCD

Refer to the response to Question 09.01.02-53.

Impact on R-COLA

There is no impact on the R-COLA.

Impact on PRA

There is no impact on the PRA.

Impact on Technical / Topical Report

There is no impact on any Technical / Topical Report.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

11/22/2013

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 1055-7184 REVISION 0**
SRP SECTION:	**09.01.02 - New and Spent Fuel Storage**
APPLICATION SECTION:	**1.9,3.8,5.1,9.1,12.3**
DATE OF RAI ISSUE:	**10/07/2013**

QUESTION NO.: 09.01.02-62:

In MUAP-13012-P (R0), Section 3.1, "Modeling Methodology," (Page 3) the paragraph states, in part, "The acceleration corresponding to the lowest natural frequency is used as the inertial amplifier to realize the seismic load effects."

The applicant is requested to address the following questions:

1. Provide a description of the mathematical model of the containment rack that is used to compute the natural frequencies.

2. The lowest natural frequency may be that of a local mode not a structural mode; therefore, the mode corresponding to the lowest natural frequency may not be the dominant mode. Therefore, provide data to show that the mode corresponding to the lowest natural frequency is the dominant structural mode.

3. Provide data to show that the contributions from higher modes are negligible.

The above information should be included in the applicable portion of the technical report.

ANSWER:

The answers to the above questions are provided as follows:

1. In accordance with SRP 3.8.4, the CR is designed and analyzed to meet the stress limits per ASME Code, Section III, Division 1, Subsection NF requirements for Class 3 component supports. This section of the Code applies to linear type supports (i.e., beam type members). Accordingly, the lowest natural frequency of the CR is calculated by treating the unsupported span (160 inches) of the CR cells between the top and bottom lateral supports as a fixed-fixed beam. The section properties of the beam are based upon the gross cross section of the CR. For the frequency calculation, the mass of the water contained inside the CR is uniformly distributed over the span length along with the dead weight. Using this mathematical model, the first mode frequency of the CR is computed to be 58 Hz.

2. As described above, the lowest natural frequency (the first mode) of the CR is calculated by treating the unsupported span (160 inches) of the CR cells between the top and

bottom lateral supports as a fixed-fixed beam. Therefore, the mode corresponding to the lowest natural frequency is the dominant structural mode during seismic events.

3. As described above, the first mode frequency of the CR is computed to be 58 Hz. Since the cut-off frequency for the floor response spectra is roughly 50 Hz, the inertial amplifier to realize the seismic load effects is equal to the zero period acceleration (ZPA). Therefore, the higher modes do not have any further influence on the seismic response of the CR since the first mode frequency is in the rigid range (more than 50 Hz).

Impact on DCD

There is no impact on the DCD.

Impact on R-COLA

There is no impact on the R-COLA.

Impact on PRA

There is no impact on the PRA.

Impact on Topical Report / Technical Report

The information described in the response will be incorporated in a future revision of technical report MUAP-13012-P.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

11/22/2013

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 1055-7184 REVISION 0**
SRP SECTION:	**09.01.02 - New and Spent Fuel Storage**
APPLICATION SECTION:	**1.9,3.8,5.1,9.1,12.3**
DATE OF RAI ISSUE:	**10/07/2013**

QUESTION NO.: 09.01.02-63:

In MUAP-13012-P (R0), Section 3.3, “Assumptions,” (Page 6) the item (1) states that “Conservatively, the fluid coupling effects surrounding the racks are neglected. The absence of fluid effects increases the rattling of the rack thereby overpredicting the stresses (or deflection) in the rack.”

The fluid coupling effects surrounding the racks should not be neglected because it may lower the natural frequency of the rack and increase the forces exerted at the rack supports. Therefore, the applicant is requested to provide technical reasons to demonstrate that neglecting the fluid coupling effect between the racks and water would be conservative for the rack design. The above information should be included in the applicable portion of the technical report.

ANSWER:

The seismic analysis of the containment rack conservatively neglected the fluid “damping” effects occurred by the water surrounding the containment rack. In MUAP-13012-P (R0), “fluid coupling effects” were used in the meaning of “fluid damping effects”. It is clear that neglecting fluid damping effect is a conservative assumption since it yields the rack displacement under earthquake.

However, MHI agrees with the NRC staff concern on the fluid coupling effect in the calculation of natural frequency and structural analysis. In MUAP-13012-P (R0),when calculating the natural frequency and structural analysis of containment rack, the mass of water contained inside the rack cells and the flux traps was already considered, but the mass of water surrounding the rack (fluid added mass) was not incorporated. That is not a conservative approach.

Therefore, the fluid coupling effects surrounding the rack (fluid added mass) will be considered in a future revision of technical report MUAP-13012-P.

Impact on DCD

There is no impact on the DCD.

Impact on R-COLA

There is no impact on the R-COLA.

Impact on PRA

There is no impact on the PRA.

Impact on Topical Report / Technical Report

The re-analysis described in the response will be incorporated in a future revision of technical report MUAP-13012-P.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

11/22/2013

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 1055-7184 REVISION 0**
SRP SECTION:	**09.01.02 - New and Spent Fuel Storage**
APPLICATION SECTION:	**1.9,3.8,5.1,9.1,12.3**
DATE OF RAI ISSUE:	**10/07/2013**

QUESTION NO.: 09.01.02-64:

In MUAP-13012-P (R0), Section 3.3, "Assumptions" (Page 6), item (2) states that "While evaluating the natural frequency of the rack, the fluid mass is considered uniformly distributed over the entire height of the rack cells thereby underestimating the natural frequency. The lower natural frequency results in larger corresponding acceleration used as seismic amplifier in the current evaluation."

The applicant is requested to address the following questions:

1. Provide the amount of fluid mass that was included in the frequency calculation.
2. The fluid mass represents only the impulsive part of liquid motion against the rack during earthquakes. Was the liquid sloshing motion against the rack considered in the analysis?
3. Provide the water depth and the dimensions of the refueling cavity.
4. The lower natural frequency may not result in larger corresponding acceleration. It depends on the shape of floor response spectra. Provide information for the floor response spectra used and the natural frequencies with and without the fluid for the rack.

The above information should be included in the applicable portion of the technical report.

ANSWER:

The answers to the above questions are provided as follows:

1. The frequency calculation accounts for the fluid mass that is contained inside the containment rack cells and the flux traps. Based on the rack dimensions, the contained fluid has a total mass of 2,634 lb.

2. Based on the methodology in TID-7024 "Nuclear Reactors and Earthquakes, (1963)" Chapter 6, the stable water height is 3/8 times the height of the fluid. For the US-APWR, the depth of the refueling cavity in the containment rack area is more than 45'-1". Then the stable water height can be calculated to be approximately 202.8 inches (16.9 feet) above the bottom of the refueling cavity. The containment rack height is approximately 197.3 inches (16.4 feet). Therefore, liquid sloshing motion does not influence against the response of the containment rack.

3. The water depth of the refueling cavity in the containment rack area during fuel assembly transport is controlled to be in the range from 45'-1" to 45'-7". The dimensions of the refueling cavity are shown in MUAP-13012-P (R0) Figures 1-1 and 1-2 (Pages 26 and 27).

4. The floor response spectra (FRS) are shown in Figures 1 through 3. In addition, the natural frequency in horizontal direction with fluid for the rack (including only inside the containment rack) is 58 Hz and the natural frequency in horizontal direction without the fluid for the rack is 81 Hz.

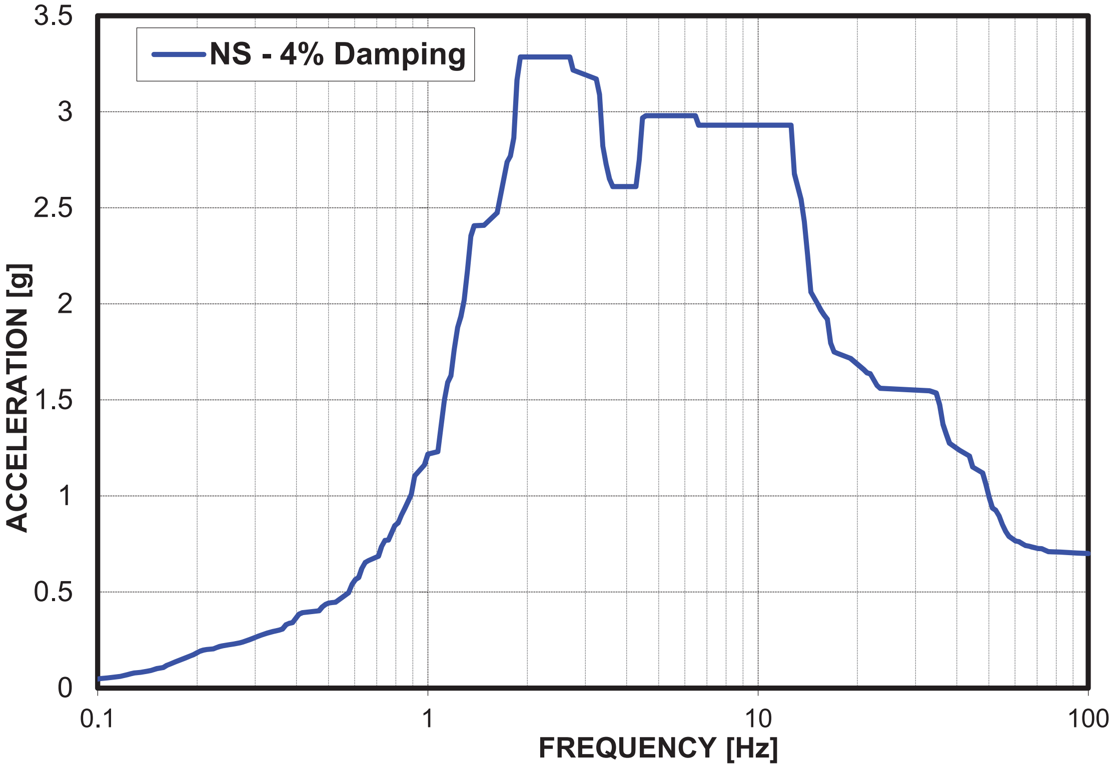


Figure 1 – FRS for Containment Rack (NS Direction)

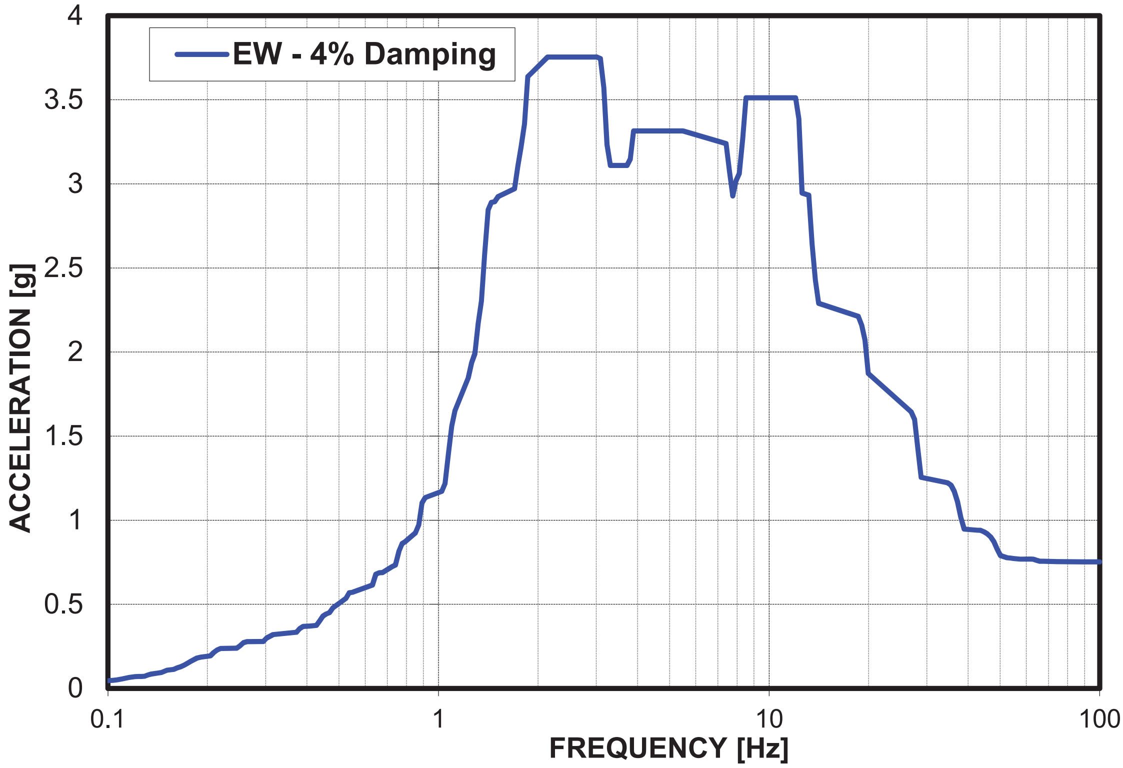


Figure 2 – FRS for Containment Rack (EW Direction)

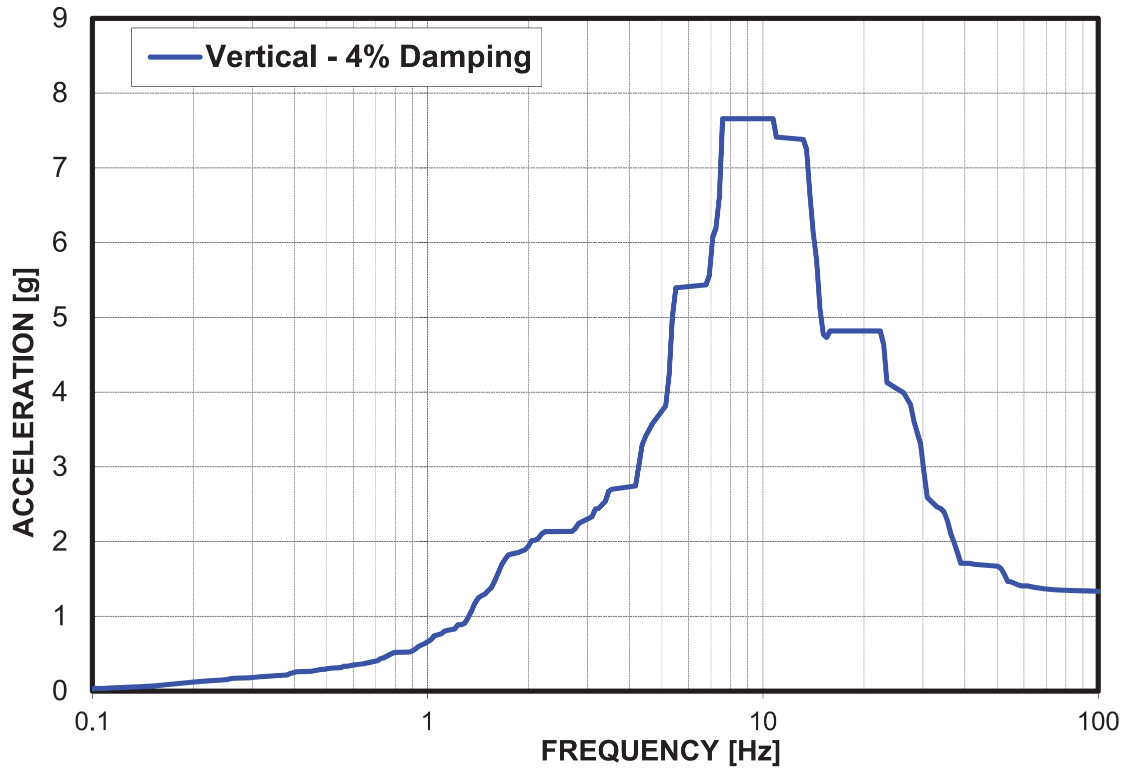


Figure 3 – FRS for Containment Rack (Vertical Direction)

Impact on DCD

There is no impact on the DCD.

Impact on R-COLA

There is no impact on the R-COLA.

Impact on PRA

There is no impact on the PRA.

Impact on Topical Report / Technical Report

The information described in the response will be incorporated in a future revision of technical report MUAP-13012-P.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

11/22/2013

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 1055-7184 REVISION 0**
SRP SECTION:	**09.01.02 - New and Spent Fuel Storage**
APPLICATION SECTION:	**1.9,3.8,5.1,9.1,12.3**
DATE OF RAI ISSUE:	**10/07/2013**

QUESTION NO.: 09.01.02-65:

In MUAP-13012-P (R0), Section 3.7.1.1 “Corresponding Acceleration,” the equation given for calculating the natural frequency of the rack is (Page 8)

fn = (22.4/2 π)*Square root((E*I*g)/(w1*L14))

Where:

w1= Load per unit length of unsupported cells (27.21 lbf/in)

The applicant is requested to provide detailed information as to how the 27.21 lbf/in is derived and whether or not w1 includes the fluid mass.

The above information should be included in the RAI response.

ANSWER:

The load per unit length of unsupported cells, w_1, accounts for the fluid mass contained inside the containment rack cells and the flux traps. The value of w_1 is calculated per the following equation:

$$w_1 = \left(\frac{M_{rack}}{L_{cell}} + \frac{M_{fluid}}{L_{cell}}\right) g$$

where M_{rack} is the mass of the containment rack (= 2,700 lb), M_{fluid} is the mass of the contained water (= 2,634 lb), L_{cell} is the length of the containment rack storage cells (= 196 in), and g is the gravitational acceleration (= 386.1 in/sec^2). The resulting value of w_1 is 10,507 lb/sec^2 (or 27.21 lbf/in).

Impact on DCD

There is no impact on the DCD.

Impact on R-COLA

There is no impact on the R-COLA.

Impact on PRA

There is no impact on the PRA.

Impact on Topical Report / Technical Report

There is no impact on the Topical Report / Technical Report.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

11/22/2013

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.: NO. 1055-7184 REVISION 0

SRP SECTION: 09.01.02 - New and Spent Fuel Storage

APPLICATION SECTION: 1.9,3.8,5.1,9.1,12.3

DATE OF RAI ISSUE: 10/07/2013

QUESTION NO.: 09.01.02-66:

In MUAP-13012-P (R0), Section 3.7.1.1 “Corresponding Acceleration,” the third and fourth paragraphs state that “The horizontal acceleration is the maximum acceleration in X-direction or Y-direction. The Floor Response Spectra (FRS) is shown in Reference 6-8. As a result, horizontal acceleration is 0.78g and vertical acceleration is 1.33g.”; and “It has been confirmed that the FRS (Reference 6-8) used in this report are larger than the one provided in Reference 6-10.”

The applicant is requested to address the following questions:

1. Does “The horizontal acceleration is the maximum acceleration in X-direction or Y-direction,” mean that the applicant only considered the larger magnitude of the acceleration of the two horizontal accelerations in the X and Y directions for the analysis of the rack? If that is the case, how does the applicant address the combined effects for the rack design due to earthquakes in the X, Y, Z components?
2. Since the staff does not have access to Reference 6.8, provide the floor response spectra for X, Y, and Z directions with descriptions including their elevations used for the rack analyses, and describe how 0.78g and 1.33g are obtained.
3. How is the natural frequency of the rack in the vertical direction calculated?
4. Explain the meaning and purpose of the statement “It has been confirmed that the FRS (Reference 6-8) used in this report are larger than the one provided in Reference 6-10.”

The above information should be included in the applicable portion of the technical report.

ANSWER:

The answers to the above questions are provided as follows:

1. The maximum horizontal acceleration in either the EW or NS direction, corresponding to the natural frequency of the containment rack, is used as the input acceleration for both the X and Y directions in the analytical model. In other words, in the analysis a 0.78g acceleration is applied in both the X and Y directions, and a 1.33g acceleration is applied in the vertical direction.

2. The floor response spectra (FRS) for the NS, EW, and vertical directions, from Reference 6-8, are shown in the response to Question No 09.01.02-64 Figures 1 through 3. The FRS data provided in Reference 6-8, are used as input to the rack analyses and 4% damping

(based on USNRC Reg. Guide 1.61 Rev. 1). The top rigger plates are installed at EL.45’10”. The rack analyses use FRS which bounds the FRS at EL.45’ 10”.

3. The stored fuel assemblies are vertically supported by the ¾” thick containment rack base plate, which in turn rests directly on the refueling cavity floor. Therefore, the natural frequency of the rack in the vertical direction is considered to be in the rigid range.

4. The FRS data provided in Reference 6-8, are used as input to the rack analyses. The final verified FRS data is documented in Reference 6-10. The purpose of the statement cited in the RAI is to confirm that the input FRS data bounds the final verified data, and therefore the analysis is conservative.

Impact on DCD

There is no impact on the DCD.

Impact on R-COLA

There is no impact on the R-COLA.

Impact on PRA

There is no impact on the PRA.

Impact on Topical Report / Technical Report

The information described in the response will be incorporated in a future revision of technical report MUAP-13012-P.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

11/22/2013

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 1055-7184 REVISION 0**
SRP SECTION:	**09.01.02 - New and Spent Fuel Storage**
APPLICATION SECTION:	**1.9,3.8,5.1,9.1,12.3**
DATE OF RAI ISSUE:	**10/07/2013**

QUESTION NO.: 09.01.02-67:

In MUAP-13012-P (R0), Section 3.7.1.2, "Fuel-to-Cell Wall Impact Loads," the first paragraph (Page 8) states that "To evaluate the impact loads between the fuel assembly and cell wall, a 2-D model of the CR plus three stored fuel assemblies has been developed using the computer code MR216 (also known as DYNARACK). Each stored fuel assembly is modeled as a lumped mass with 2 translational degrees of freedom in the horizontal x and y directions. The CRs are modeled as a separate body with 3 degrees of freedom (2 translational, 1 rotational). Each fuel mass is surrounded by 4 gap elements (in the +x, +y, -x, and -y directions), which track the impacts between the fuel and the CR during the seismic event. This model includes the effects of fluid coupling between the fuel assemblies and the cell walls."

The applicant is requested to address the following questions:

1. Provide description for the mathematical model of the rack and its boundary conditions and the earthquake input motions to the model.
2. Is the performed impact analysis a linear or nonlinear structural analysis? If it is a linear analysis, the applicant is requested to provide technical details for how this analysis is carried out; if it is a nonlinear analysis, according to SRP 3.7.1, more than four time histories should be used. Provide and describe the number of time histories used in the analysis.
3. Provide a sketch for the 2-D model and describe how the fuel mass is calculated.
4. When subjected to the horizontal seismic motion, the deflection of the containment rack is not uniform along its height because the containment rack is anchored to the adjacent wall only at its top and bottom. Also due to the effect of soil-structure interaction, the reactor complex experiences a rocking motion at its base; as a result of this, the motion at the base of the containment rack is different from that at the top of the rack. A 2-D model in x-y plane of the containment rack may not be appropriate. Provide technical rationale to justify that the 2-D model is appropriate, and why a 3-D model is not required.

The above information should be included in the applicable portion of the technical report.

ANSWER:

The impact analysis was a nonlinear structure analysis and used a 2-D model. In addition, this analysis used one time history, not in accordance with SRP 3.7.1.

In SRP 3.8.4 appendix D, Section 3, Seismic and Impact Loads, the fifth paragraph states that, “Because of gaps between fuel assemblies and the walls of the guide tubes, additional loads will be generated by the impact of fuel assemblies during a postulated seismic excitation. Additional loads resulting from this impact effect may be determined by estimation the kinetic energy of fuel assembly.” The maximum velocity of the fuel assembly may be estimated to be the spectral velocity associated with the natural frequency of the submerged fuel assembly.

The methodology for fuel-to-cell impact analysis will be changed from a nonlinear analysis to a linear analysis considering the kinetic energy of fuel assembly in accordance with SRP 3.8.4 Appendix D. Therefore, the impact analysis will be performed again in a future revision of technical report MUAP-13012-P.

Impact on DCD

There is no impact on the DCD.

Impact on R-COLA

There is no impact on the R-COLA.

Impact on PRA

There is no impact on the PRA.

Impact on Topical Report / Technical Report

The change described in the response will be incorporated in a future revision of technical report MUAP-13012-P.

11/22/2013

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 1055-7184 REVISION 0**
SRP SECTION:	**09.01.02 - New and Spent Fuel Storage**
APPLICATION SECTION:	**1.9,3.8,5.1,9.1,12.3**
DATE OF RAI ISSUE:	**10/07/2013**

QUESTION NO.: 09.01.02-68:

In MUAP-13012-P (R0)), Section 3.7.1.2, “Fuel-to-Cell Wall Impact Loads”, the second paragraph (Page 8) states that “The maximum fuel-to-cell impact load calculated by this model is 4,628 lbf shown in Table 3-4 (Reference 6-11). The maximum allowable impact load on the cell membrane, considering an additional safety factor of 2, based on the yield strength is then given as 24,994 lbf.”

The applicant is requested to address the following questions:

1. How is the allowable impact load of 24,994 lbf on the cell membrane obtained?
2. What is the allowable impact load for the fuel assembly? The applicant needs to show that the fuel assembly is not damaged by the impact loads.

The above information should be included in the RAI response.

ANSWER:

The answers to the above questions are provided as follows:

2. The 4,627 lbf maximum impact load is less than both of the above calculated loads. Based on the 24,994 lbf impact load allowable determined above, the safety factor against cell membrane failure is 5.40. Therefore, the stresses in the cell membrane are acceptable for the postulated fuel impacts.

   As described in the response to Question No. 09.01.02-67 of this RAI, the analysis model of the impact analysis will be changed in a future revision of technical report MUAP-13012-P.

   The allowable impact load for the fuel assembly will be shown in a future revision of technical report MUAP-13012-P. The fuel assembly is not damaged by the impact loads as will be shown in a future revision of technical report MUAP-13012-P.

Impact on DCD

There is no impact on the DCD.

Impact on R-COLA

There is no impact on the R-COLA.

Impact on PRA

There is no impact on the PRA.

Impact on Topical Report / Technical Report

There is no impact on the Topical Report / Technical Report.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

11/22/2013

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 1055-7184 REVISION 0**
SRP SECTION:	**09.01.02 - New and Spent Fuel Storage**
APPLICATION SECTION:	**1.9,3.8,5.1,9.1,12.3**
DATE OF RAI ISSUE:	**10/07/2013**

QUESTION NO.: 09.01.02-69:

In MUAP-13012-P (R0), Section 3.7.2, "Rack Stress Evaluation," the paragraph (Page 9) states that "Regarding supports of the fuel cell (see Figure 2-1), the net section maximum bending moments, shear forces and tensile stress can also be determined at the mounting angle. Based on these, the maximum stress in the CR can be evaluated at the mounting angle. The evaluation of the fuel cell and mounting angle is described below. In addition, the results at other points, such as the end rigger plates, are shown in Tables 3-5 and 3-6."

The size and dimensions of the mounting angle and rigger plates are not given in Figure 2-1 or in Tables 3-5 and 3-6. The applicant is requested to provide this information. The above information should be included in the applicable portion of the technical report.

ANSWER:

The size and dimensions of the mounting angle and rigger plates are shown in Figure 1 and provided below.

The size and dimensions of the mounting angle and rigger plates will be shown in a future revision of technical report MUAP-13012-P.

Dimensions	
Rigger Plate Width	26 in
Rigger Plate Thickness	0.5 in
Mounting Angle Depth	24 in
Mounting Angle Width	5 in
Mounting Angle Thickness	0.5 in
Mounting Angle Gusset Width	4 in
Mounting Angle Gusset Thickness	0.25 in

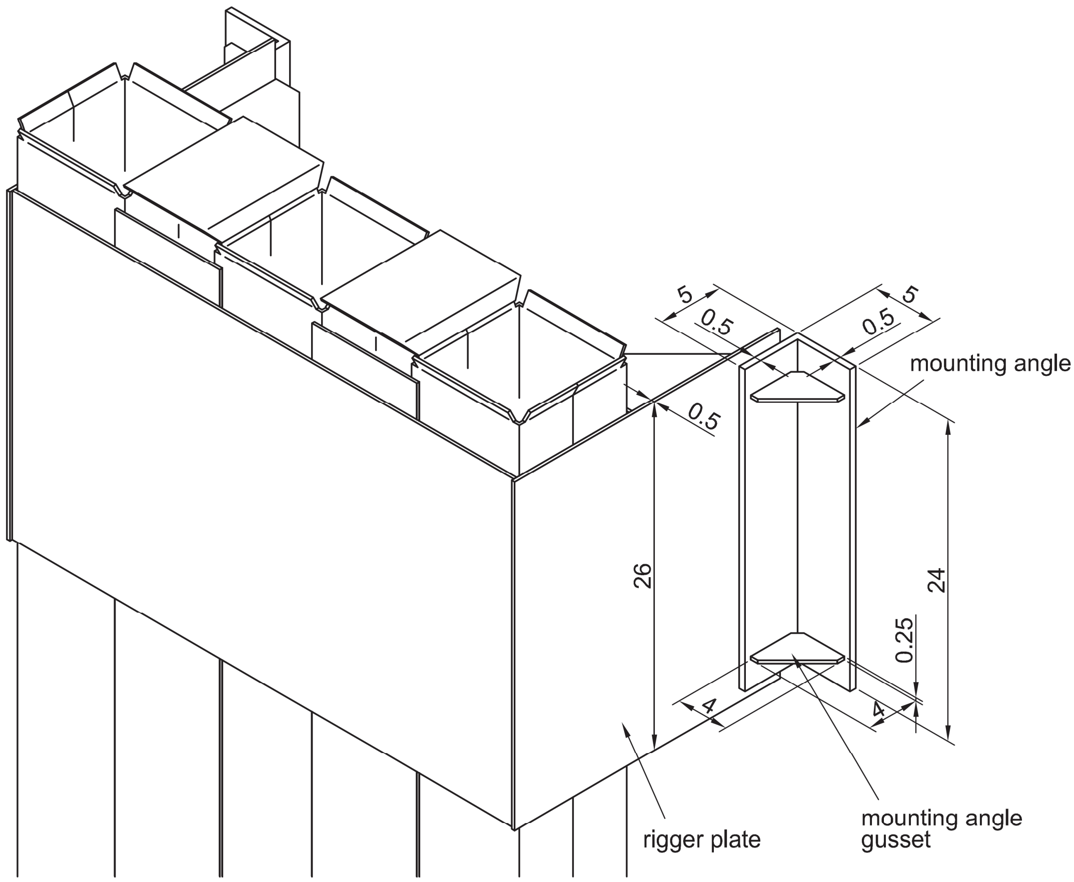


Figure1 Outline drawing of mounting angle and rigger plate

Impact on DCD

There is no impact on the DCD.

Impact on R-COLA

There is no impact on the R-COLA.

Impact on PRA

There is no impact on the PRA.

Impact on Topical Report / Technical Report

The information described in the response will be incorporated in a future revision of technical report MUAP-13012-P.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

11/22/2013

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 1055-7184 REVISION 0**
SRP SECTION:	**09.01.02 - New and Spent Fuel Storage**
APPLICATION SECTION:	**1.9,3.8,5.1,9.1,12.3**
DATE OF RAI ISSUE:	**10/07/2013**

QUESTION NO.: 09.01.02-70:

In MUAP-13012-P (R0), Section 3.7.2.1, “Fuel Cell,” the bending stress is given by the equation on page 9:

Bending stress = (M1/Z1) + (M2/Z2) = 1761 psi

Where:

Z1 : X-axis section modulus of the rack (211.9 in3)
Z2 : Y-axis section modulus of the rack (95.95 in3)

The applicant is requested to address the following questions:

1. Describe how M1 and M2 are calculated.
2. Provide dimensions/sketches of the cross section of the fuel cell.
3. Why are Z1 and Z2 the moduli of the rack and not moduli of the fuel cell since the calculation pertains to the fuel cell?

The above information should be included in the applicable portion of the technical report.

ANSWER:

The answers to the above questions are provided as follows:

1. The method of calculating M_1 and M_2 is provided below. The mathematical model is shown in Figure 1.

2. The dimensions/sketches of cross section of the fuel cell will be provided in Figure 2.

3. In Section 3.7.2.1, “Fuel Cell”, the bending stress considering rack cell and rack filler panel are evaluated (as shown in Figure 2).

   Therefore, the moduli of the rack (Z1, Z2) are used in the calculation of the bending stress.

$$M_1 = \frac{W_y \times L_1}{12}$$

$$M_2 = \frac{W_x \times L_1}{12}$$

$$W_x = W_y = M_{total} \times \max(Z_x, Z_y)$$

Where:

M_1 : X-axis bending moment

M_2 : Y-axis bending moment

W_x : Seismic load in X direction

W_y : Seismic load in Y direction

M_{total} : Total mass accelerated during a SSE event

Z_x : Corresponding acceleration in X-direction

Z_y : Corresponding acceleration in Y-direction

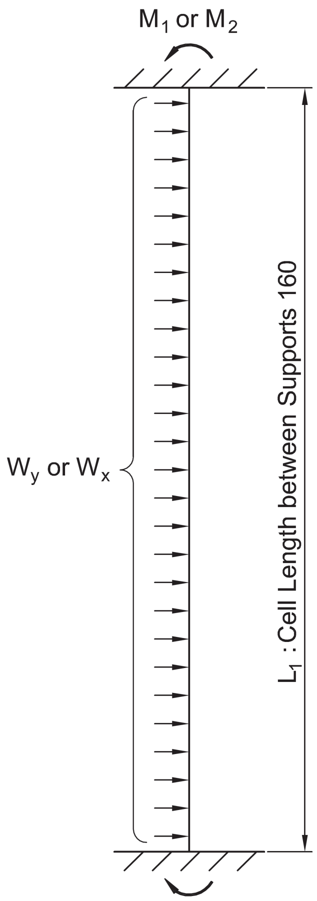


Figure 1 Sketch showing mathematical model

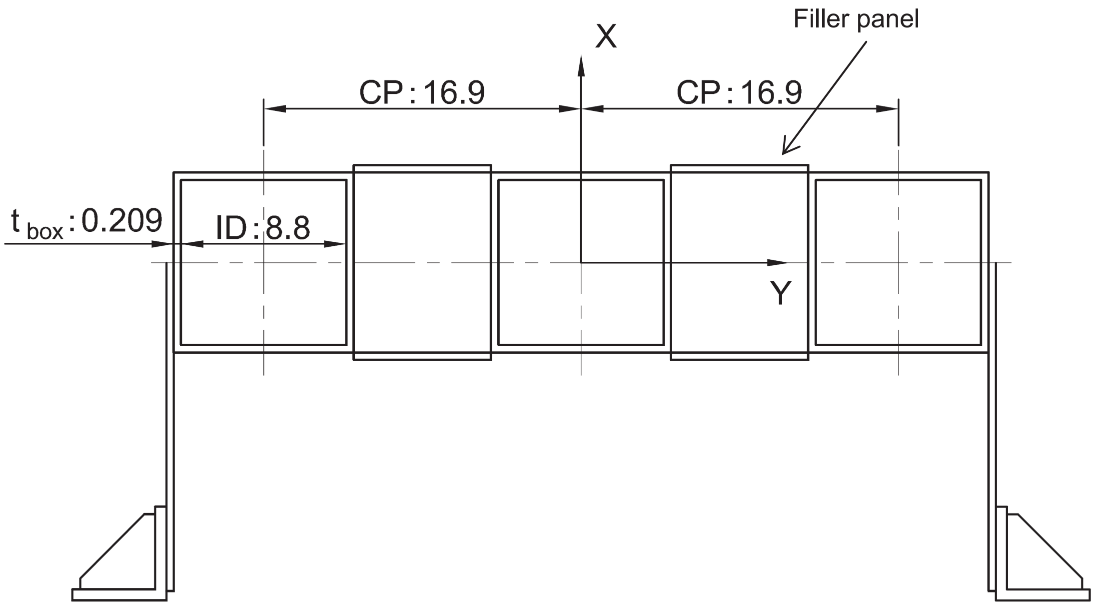


Figure 2 Sketch showing dimensions of the cross section of the fuel cell including filler panel

Impact on DCD

There is no impact on the DCD.

Impact on R-COLA

There is no impact on the R-COLA.

Impact on PRA

There is no impact on the PRA.

Impact on Topical Report / Technical Report

The information described in the response will be incorporated in a future revision of technical report MUAP-13012-P.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

11/22/2013

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.: **NO. 1055-7184 REVISION 0**

SRP SECTION: **09.01.02 - New and Spent Fuel Storage**

APPLICATION SECTION: **1.9,3.8,5.1,9.1,12.3**

DATE OF RAI ISSUE: **10/07/2013**

QUESTION NO.: 09.01.02-71:

In MUAP-13012-P (R0), Section 3.7.2.2, “Mounting Angle Tensile and Bending,” (Pages 9 and 10) the paragraph states the following:

W2 : Reaction at limiting cross section (6,168 lbf)
A1 : Area of the mounting angle limiting cross section (13.62 in^2)
M3 : Moment about VT-axis (44,440 lbf in)
M4 : Moment about Y-axis (58,160 lbf in)
Z3 : VT-axis section modulus of the rack (2.047 in^3)
Z4 : Y-axis section modulus of the rack (44.31 in^3)

The applicant is requested to address the following questions:

1. Provide details/sketches to show how these quantities are calculated.
2. Both Z4 in Section 3.7.2.2 and Z2 in Subsection 3.7.2.1 represent the Y-axis section modulus of the rack. Why Z4 has a value of 44.31 in^3, which is different from Z2 that has a value of 95.95 in^3?

The requested information should be included in the applicable portion of the technical report.

ANSWER:

The answers to the above questions are provided as follows:

1. The details/sketches are shown in Figures 1 and 2. The methods of calculating these quantities are provided below.

2. Z_2 shows the Y-axis section modulus of the rack, which includes fuel cell and filler panel. Z_4 shows the Y-axis section modulus of the mounting angle, which includes mounting angle and mounting angle gusset.

   The explanatory note of Z_4 shown in above description will be added in a future revision of technical report MUAP-13012-P.

Detailed calculation:

$$W_2 = (\frac{W_x}{2} + \frac{W_y}{2}) \times \sin 45°$$

$$A_1 = d_{ma} \times t_{ma} + 2 \times w_{g\_plate4} \times t_{g\_p2}$$

$$M_3 = (\frac{W_y}{2}) \times \cos 45° \times \frac{L_{am\_ma2}}{2}$$

$$M_4 = \frac{M_2}{2}$$

$$Z_3 = \frac{I_{ma\_vt2}}{t_{ma} + w_{g\_plate4} - C_{y\_ma2}}$$

$$Z_4 = \frac{I_{ma\_y2}}{\frac{w_p}{2}}$$

Where :

W_2 : Reaction at limiting cross section

W_x : Seismic load in X direction

W_y : Seismic load in Y direction

A_1 : Area of the mounting angle limiting cross section

M_3 : Moment about VT-axis

Z_3 : VT-axis section modulus of the mounting angle including mounting angle and mounting angle gusset.

I_{ma_vt2} : Moment of inertia of the mounting angle including mounting angle and mounting angle gusset.

M_4 : Moment about Y-axis

M_2 : Y-axis bending moment

Z_4 : Y-axis section modulus of the mounting angle including mounting angle and mounting angle gusset.

I_{ma_y2} : Moment of inertia of the composite section at limiting cross section

Dimensions:

d_{ma} : Mounting angle depth (24 in)

t_{ma} : Mounting angle thickness (0.5 in)

w_{g_plate4} : The width of the mounting angle gusset along the 45°section line (3.248 in)

t_{g_p2} : Mounting angle gusset thickness (0.25 in)

w_{g_plate3} : Mounting angle gusset width (4 in)

$w_{g\_plate31}$: Width of mounting angle gusset at the end ($\frac{19}{32}$ in)

L_{am_ma2} : Moment arm (28.81 in)

c_{y_ma2} : Center of gravity of the composite section at limiting cross section (0.473 in)

w_p : Width of top and bottom support panels (26 in)

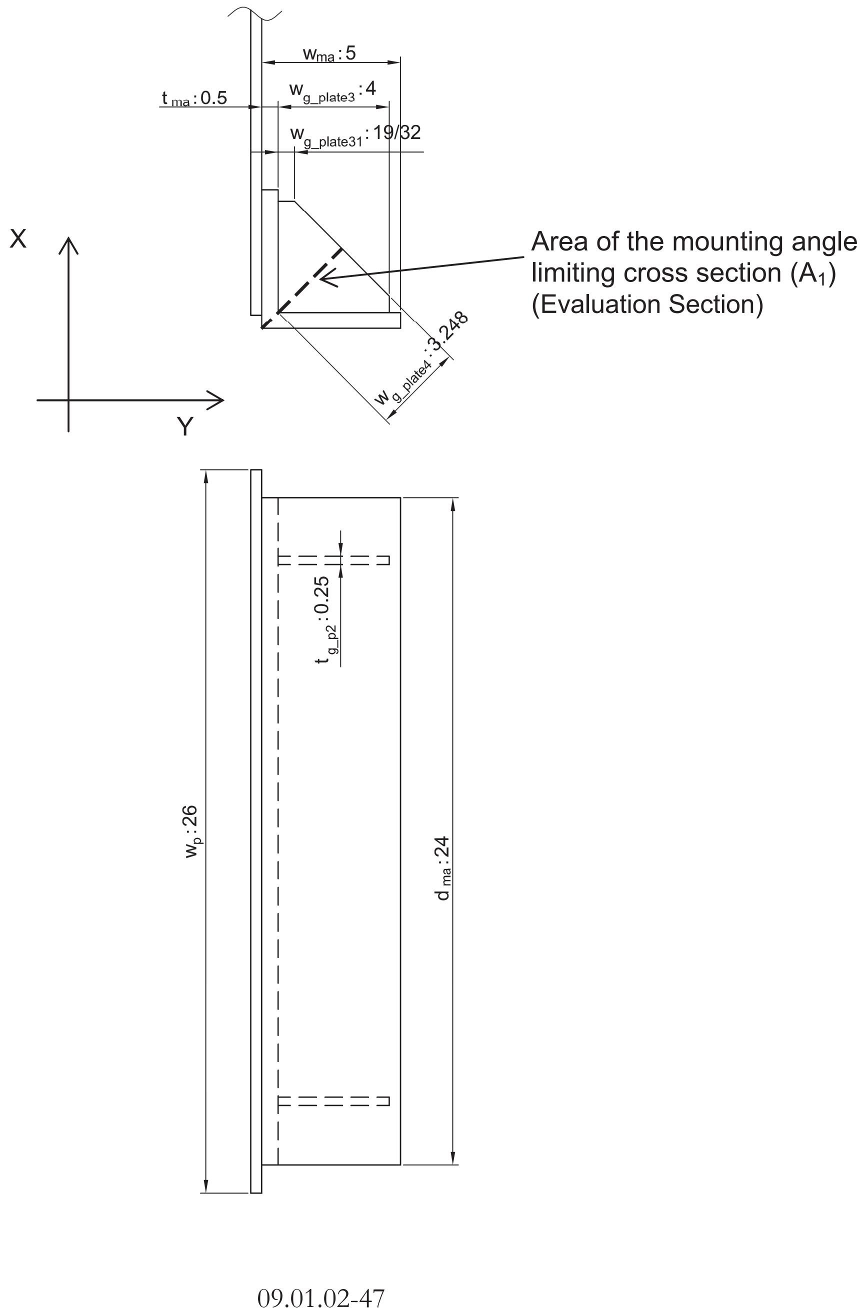


Figure 1 Sketch showing dimensions

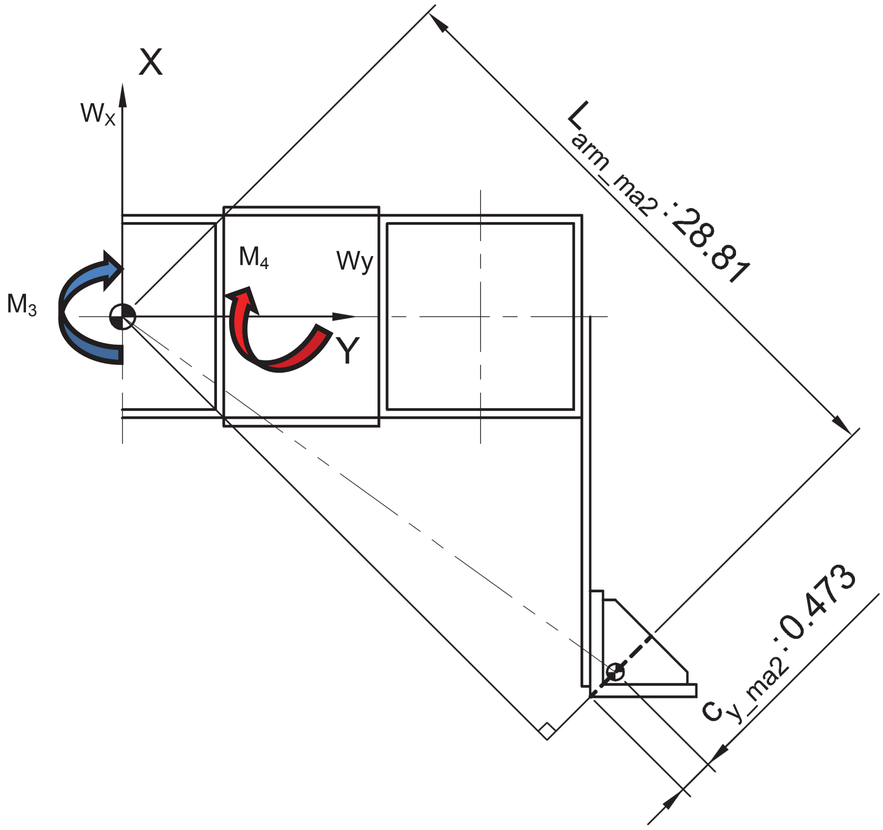


Figure 2 Sketch showing center of gravity

Impact on DCD

There is no impact on the DCD.

Impact on R-COLA

There is no impact on the R-COLA.

Impact on PRA

There is no impact on the PRA.

Impact on Topical Report / Technical Report

The information described in the response will be incorporated in a future revision of technical report MUAP-13012-P.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

11/22/2013

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 1055-7184 REVISION 0**
SRP SECTION:	**09.01.02 - New and Spent Fuel Storage**
APPLICATION SECTION:	**1.9,3.8,5.1,9.1,12.3**
DATE OF RAI ISSUE:	**10/07/2013**

QUESTION NO.: 09.01.02-72:

In MUAP-13012-P (R0), Section 3.7.2.3, “Mounting Angle Shear Stress,” (Page 10) the first paragraph states that “The moment about X-axis is resisted by a force couple on the pair of mounting angles on opposite sides of the CR. This force couple causes a shear stress to develop in the mounting angles. This shear stress is combined with the shear stress caused by the load in the Y direction and the VT-direction acceleration in the following calculation.”

The applicant is requested to provide a sketch that shows how the resulting shear was formed for the CR and how the resisting shear was provided by the mounting angle. The requested information should be included in the RAI response.

ANSWER:

The sketch that provides how the shear was formed for the CR is shown in Figure 1.

The methodology for calculating the resulting shear (W_3) at evaluation section (shown in Figure 1) is provided below. Additionally, the resisting shear (F_{v1}) by a force couple is provided below.

Detailed calculation:

$$W_3 = \sqrt{(F_{v1} + F_{v3})^2 + F_{v2}{}^2}$$

$$A_2 = d_{ma} \times t_{ma} + 2 \times w_{g\_plate3} \times t_{g\_p2}$$

$$F_{v1} = \frac{M_1}{L_{span}}$$

$$F_{v2} = \frac{W_y}{4}$$

$$F_{v3} = (Z_{vt} - 1) \times \frac{M_{total}}{4}$$

Where :

W_3 : Combined shear force on one mounting angle (resulting shear force)

A_2 : Area of the mounting angle

F_{v1} : Shear force on mounting angle due to moment about X-axis (the resisting shear)

F_{v2} : Shear force on one mounting angle due to seismic load in Y-direction

F_{v3} : Shear force on one mounting angle due to vertical acceleration

M_1 : X-axis bending moment

W_y : Seismic load in Y direction

Z_{vt} : Acceleration in Vertical-direction

M_{total} : Total mass accelerated during a SSE event

(M_{total} includes:.Mass of rack, mass of fuel assembly, and mass of water in the rack.)

Dimensions:

d_{ma} : Mounting angle depth (24 in)

t_{ma} : Mounting angle thickness (0.5 in)

w_{g_plate3} : Mounting angle gusset width (4 in)

t_{g_p2} : Mounting angle gusset thickness (0.25 in)

L_{span} : Horizontal span between mounting angle (43.5 in)

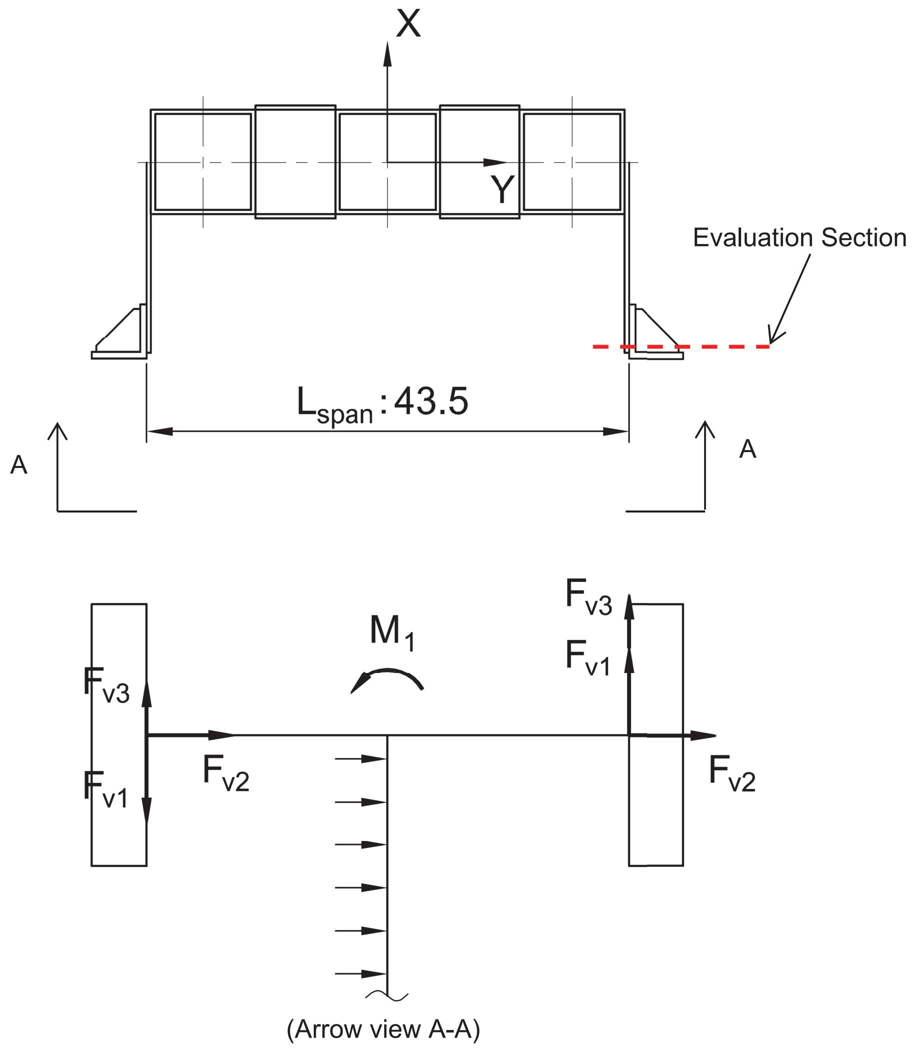


Figure 1 A sketch showing the shear stress

Impact on DCD

There is no impact on the DCD.

Impact on R-COLA

There is no impact on the R-COLA.

Impact on PRA

There is no impact on the PRA.

Impact on Topical Report / Technical Report

There is no impact on the Topical Report / Technical Report.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

11/22/2013

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 1055-7184 REVISION 0**
SRP SECTION:	**09.01.02 - New and Spent Fuel Storage**
APPLICATION SECTION:	**1.9,3.8,5.1,9.1,12.3**
DATE OF RAI ISSUE:	**10/07/2013**

QUESTION NO.: 09.01.02-73:

In MUAP-13012-P (R0), Section 3.7.2.4, “Mounting Angle-to-Wall Welds,” (Page 11) the paragraph states that “The angle plates may be welded directly to the wall by 3/16 inches fillet welds. The stress in the anchor weld is calculated below.”

Stress in anchor = W4/A3 = 1052 psi

Where:

W4 : Combined shear force on one mounting angle (4,738 lbf)
A3 : Area of weld between mounting angle and wall (4.508 in2)

The applicant is requested to provide details/sketches as to how W4 and A3 are calculated and the type of weld material used in fillet weld. The requested information should be included in the RAI response.

ANSWER:

A sketch is shown in Figure 1 and the answers to the W_4 and A_3 are provided as follows.

The material corresponding to SA240 will be used as the welding material.

Detailed calculation

$$W_4 = \sqrt{(\frac{W_X}{4})^2 + F_{v2}{}^2 + (F_{v1} + F_{v3})^2}$$

$$A_3 = (d_{ma} + 2 \times w_{ma}) \times \frac{w_{ap\_wall}}{\sqrt{2}}$$

Where :

W_4 : Combined shear force on one mounting angle

W_x : Seismic load in X direction

F_{v1} : Shear force on mounting angle due to moment about X-axis

F_{v2} : Shear force on one mounting angle due to seismic load in Y-direction

F_{v3} : Shear force on one mounting angle due to vertical acceleration

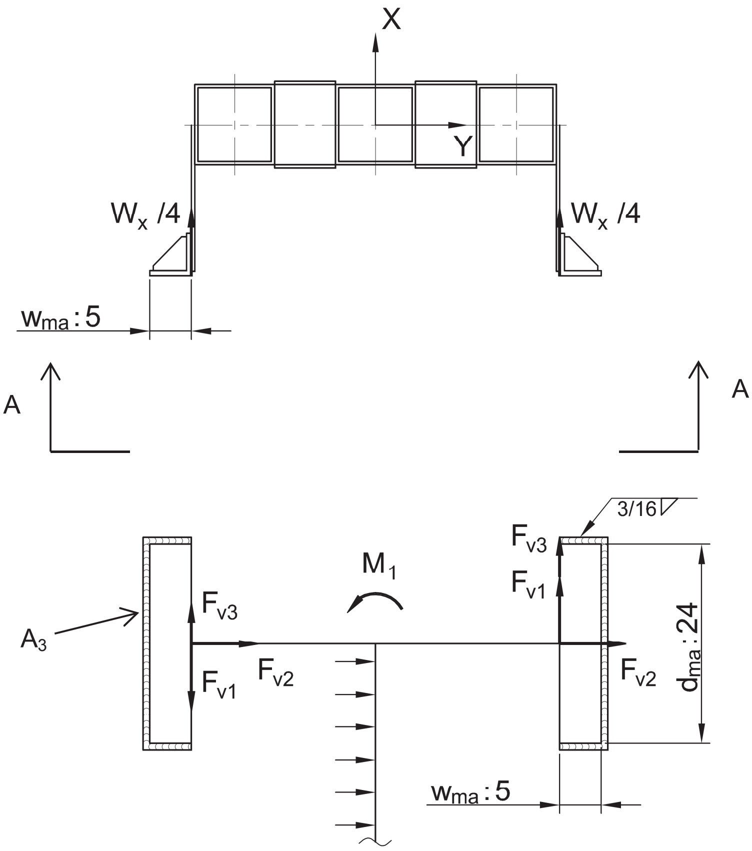
A_3 : Area of weld between mounting angle and wall

Dimensions

d_{ma} : Mounting angle depth (24 in)

w_{ma} : Mounting angle width (5 in)

w_{ap_wall} : Weld size between mounting angle and wall ($\frac{3}{16}$ in)



(Arrow view A-A)

Figure 1 Sketch showing Mounting Angle-to-Wall Welds

Impact on DCD

There is no impact on the DCD.

Impact on R-COLA

There is no impact on the R-COLA.

Impact on PRA

There is no impact on the PRA.

Impact on Topical Report / Technical Report

There is no impact on the Topical Report / Technical Report.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

11/22/2013

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 1055-7184 REVISION 0**
SRP SECTION:	**09.01.02 - New and Spent Fuel Storage**
APPLICATION SECTION:	**1.9,3.8,5.1,9.1,12.3**
DATE OF RAI ISSUE:	**10/07/2013**

QUESTION NO.: 09.01.02-75:

In MUAP-13012-P (R0), Section 4.3.1, “Straight Shallow Drop Event” (Page 15), the applicant states that the straight shallow drop event does not compromise the structural integrity of the CR. The applicant, however, does not address the structural integrity of the fuel assembly.

The applicant is requested to address the structural integrity of the fuel assembly. Also, the applicant is requested to provide information on how the design limit of 6.5 inches for the CR cellular structure plastic deformation and the allowable weld stress of 35,690 psi mentioned in this section are obtained. The above information should be included in the applicable portion of the technical report.

ANSWER:

The structural integrity of the dropped fuel assembly is not explicitly evaluated since the design basis radiological calculations for the plant consider the consequences of a fuel accident event resulting in 100% rupture of all fuel rods in one fuel assembly.

The design limit of 6.5 inches for the CR cellular structure plastic deformation is equal to the distance from the top of the CR cell structure to the top of a stored fuel assembly inside the CR. In other words, plastic deformation within the top 6.5 inches of the CR cellular structure poses no risk of a critical fuel event.

ASME Code, Section III, Division 1, Subsection NF (2001 Edition through 2003 Addenda) does not specify an allowable weld stress for Level D conditions. Therefore, an appropriate limit for the allowable weld stress is determined as follows:

$$F_w = (0.3S_u) \times Factor$$

where:

$0.3S_u$	is the allowable weld stress for Level A Service Conditions (S_u = 66,100 psi for SA-240 304L at 200ºF);
Factor	is the ratio of the allowable base metal shear stress limits for Level D and Level A Service Conditions (= $0.72S_y$ / $0.4S_y$ = 1.8).

The resulting value for F_w is 35,694 psi.

Impact on DCD

There is no impact on the DCD.

Impact on R-COLA

There is no impact on the R-COLA.

Impact on PRA

There is no impact on the PRA.

Impact on Topical Report / Technical Report

The information described in the response will be incorporated in a future revision of technical report MUAP-13012-P.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

11/22/2013

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 1055-7184 REVISION 0**
SRP SECTION:	**09.01.02 - New and Spent Fuel Storage**
APPLICATION SECTION:	**1.9,3.8,5.1,9.1,12.3**
DATE OF RAI ISSUE:	**10/07/2013**

QUESTION NO.: 09.01.02-76:

In MUAP-13012-P (R0), Section 4.3.2, “Stuck Fuel Event,” (Page 16) the paragraph states that “Result of the analysis show that the maximum stress in the rack cell due to a stuck fuel assembly is only 1,197 psi, which is well below the material yield stress. Therefore, the CR are adequate to withstand a 4,400 lbf uplift force due to a stuck fuel assembly.”

The applicant is requested to provide the following information:

1. A description of the mathematical model and assumptions used in this analysis.
2. How is the 4,400 lbf uplift force determined to be enough to uplift the stuck fuel?
3. The location of the maximum stress of 1,197 psi. Does the location of the maximum stress depend on the assumption of how the fuel assembly gets stuck?
4. Are the mounting angles and welds adequate for the 4,400 lbf uplifting force?
5. What is the maximum stress developed in the fuel assembly? The applicant is requested to show that the integrity of the fuel assembly is maintained.

The above information should be included in the applicable portion of the technical report.

ANSWER:

The answers to the above questions are provided below.

1. The stuck fuel event is analyzed by manual calculations using strength of materials formula. Specifically, the following scenarios are analyzed:
   a. The entire 4,400 lbf uplift force is resisted by a single storage cell through direct tension (only 50% of cell area is credited, shown in Figure 1).
   b. The 4,400 lbf uplift force is applied at a rack corner location causing axial tension and two plane bending on the rack cross section (the mounting angles are ignored and the rack cell structure is treated as a cantilevered beam secured at its base, shown in Figure 2).

2. In DCD Revision 4, Section 9.1.2.2.4 “New Fuel Storage Rack, Spent Fuel Storage Rack and Containment Rack Design” (Page 9.1-12) the third paragraph states, “Uplift force analysis is also performed for new fuel racks, spent fuel racks and containment racks design, and described in the technical reports (Refs. 9.1.7-8 and 9.1.7-32). 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."

3. The maximum stress of 1,197 psi is based on an assumption that the 4,400 lbf uplift force is resisted over an area that is equal to 50% of the cross sectional area of a single storage cell. The low calculated stress, as compared to the yield strength of the material (21,400 psi), indicates a large margin of safety against overstress even if the assumed loaded area is significantly reduced.

4. The stress induced in the mounting angles and their connecting welds due to the stuck fuel assembly condition are bounded by the calculated stresses under seismic conditions. This is because the mounting angles and their connecting welds are analyzed for a net vertical (upward) acceleration of 0.33g (= 1.33g vertical acceleration - 1g gravity), which is applied conservatively to the total combined mass of the rack, the stored fuel assemblies, and the contained water (11,184 lb). Thus, the seismic analysis considers a vertical uplift force of 0.33 x 11,184 lb = 3,690 lb. By comparison, the net uplift force on the mounting angles and their connecting welds due to a stuck fuel assembly is 4,400 lb – 0.87 x 2,700 lb (buoyant weight of rack) = 2,051 lb.

5. It is shown in MUAP-07016-P(R4) Section 4.2 that the fuel assembly will not experience excessive deformation under four times of gravity (4g) in axial direction. Therefore, since 4g axial load is larger than the uplift force, it is considered that the integrity of the fuel assembly is maintained under the uplift force.

Tension caused by Uplift Force

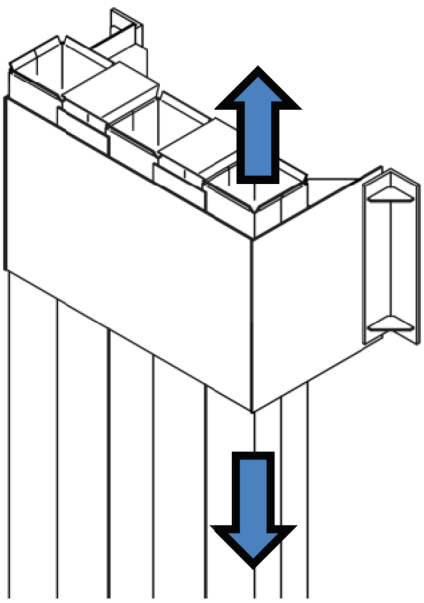


Figure 1 Sketch of tension caused by Uplift Force

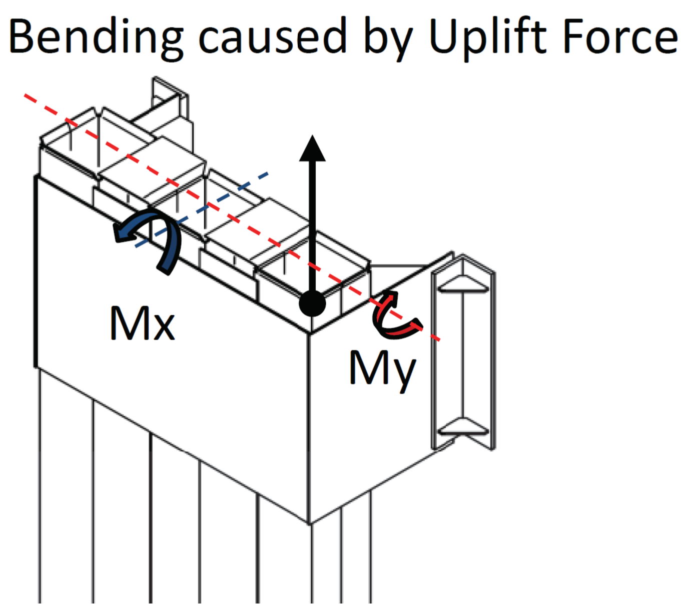


Figure 2 Sketch of Bending caused by Uplift Force

Impact on DCD

There is no impact on the DCD.

Impact on R-COLA

There is no impact on the R-COLA.

Impact on PRA

There is no impact on the PRA.

Impact on Topical Report / Technical Report

The information described in the response will be incorporated in a future revision of technical report MUAP-13012-P.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

11/22/2013

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.: **NO. 1055-7184 REVISION 0**

SRP SECTION: **09.01.02 - New and Spent Fuel Storage**

APPLICATION SECTION: **1.9,3.8,5.1,9.1,12.3**

DATE OF RAI ISSUE: **10/07/2013**

QUESTION NO.: 09.01.02-77:

In DCD Revision 3, Mark-up Section 9.1.2.2.4, “New Fuel Storage Rack, Spent Fuel Storage Rack and Containment Rack Design,” (Page 9.1-11) the first paragraph states, “Structural design and stress analysis of the new fuel storage racks, spent fuel storage racks and containment racks are evaluated in accordance with the seismic Category I requirements of Regulatory Guide 1.29.”

The staff noticed that Regulatory Guide 1.29, “Seismic Design Classification,” does not contain the structural design and stress analysis requirements for seismic Category I structures. The applicant is requested to clarify that the reference RG 1.29 is for classification and to provide the evaluation criteria for the new fuel racks, the spent fuel racks, and the containment racks. The applicant response should be included in the applicable portion of the DCD.

ANSWER:

The Regulatory Guide 1.29 is for Seismic design classification and does not contain the structural design and stress analysis requirements for seismic Category I structures. The evaluation criteria for the new fuel racks, the spent fuel racks, and the containment racks are described in DCD Revision 4, Section 9.1.2.1 “Design Bases”, fourth paragraph (Page 9.1-6) wherein it is stated that “The requirements of ASME Code Section III, Division I, Article NF3000 are used as the criteria for evaluation of stress analysis.”

Impact on DCD

There is no impact on the DCD.

Impact on R-COLA

There is no impact on the R-COLA.

Impact on PRA

There is no impact on the PRA.

Impact on Topical Report / Technical Report

There is no impact on the Topical Report / Technical Report.

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 Pit 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 normal conditions is less than 1.0 DCD_09.01.02-52 without credit for the soluble boron, and less than 0.95 with partial credit taken for soluble boron. 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. New fuel storage racks, spent fuel storage racks, and containment racks are seismic category I components.

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.

Refueling cavity water level instrumentation annunciates in the MCR for refueling cavity low water level. A portable area radiation monitor that alarms locally is provided in the refueling cavity area near the containment racks for personnel protection and general surveillance when fuel is temporarily stored. The Containment HVAC system removes radioactivity from the atmosphere above the refueling cavity and discharges through the plant vent. The ventilation system is continuously monitored by gaseous radiation monitors. If radiation levels reach a predetermined setpoint, an alarm actuates in the MCR. DCD_09.01.02-56

See Chapters 7, 9, and 12 for details.

3.1.6.5 Criterion 64 – Monitoring Radioactivity Releases

> Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of LOCA fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents

3.1.6.5.1 Discussion

The containment atmosphere is continuously monitored during normal and transient station operations, using the containment particulate, and gaseous radiation monitors. Under accident conditions, samples of the containment atmosphere provide data on existing airborne radioactive concentrations within the containment.

Radioactivity levels contained in the facility effluent and discharge paths and in the plant environs are continuously monitored during normal and accident conditions by the plant Radiation Monitoring Systems.

Portable radiation detection instruments are provided to periodically monitor radiation levels in the R/B spaces, which contain components for recirculation of LOCA fluids, the refueling cavity area near the containment racks when fuel is temporarily stored, and in the A/B for components that process radioactive wastes. In addition to the installed detectors, periodic plant environmental surveillance is established. Measurement capability and reporting of effluents are based on the guidelines of RG 1.183 (Reference 3.1-7) and RG 1.21 (Reference 3.1-17). DCD_09.01.02-56

Radiation Monitoring Systems are discussed in Chapter 11, Section 11.5, and Chapter 12, Section 12.3.

3.1.7 Combined License Information

COL 3.1(1) *The COL Applicant is to provide a design that allows for the appropriate inspections and layout features of the ESWS.*

3.1.8 References

3.1-1 Domestic Licensing of Production and Utilization Facilities, General Design Criteria for Nuclear Power Plants, Energy. Title 10, Code of Federal Regulations, Part 50, Appendix A, U.S. Nuclear Regulatory Commission, Washington, DC.

3.1-2 Codes and Standards, Domestic Licensing of Production and Utilization Facilities, Energy. Title 10, Code of Federal Regulations, Part 50.55a, U.S. Nuclear Regulatory Commission, Washington, DC.

3.1-3 Fire Protection Program, Auxiliary Systems, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plant. NUREG-0800, SRP 9.5.1, Rev. 5, U.S. Nuclear Regulatory Commission, Washington, DC, March 2007.

3.1-4 Fire Protection for Nuclear Power Plants. Regulatory Guide 1.189, Rev. 1, U.S. Nuclear Regulatory Commission, Washington, DC, March 2007.

3.1-5 Rules for Construction of Nuclear Power Plant Components, ASME Boiler and Pressure Vessel Code, Section III, 1992 Edition through the 1992 Addenda. American Society of Mechanical Engineers.

3.1-6 Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors. Regulatory Guide 1.183, U.S. Nuclear Regulatory Commission, Washington, DC, July 2000.

3.1-7 Domestic Licensing of Production and Utilization Facilities, Fracture Toughness Requirements, Energy. Title 10, Code of Federal Regulations, Part 50, Appendix G, U.S. Nuclear Regulatory Commission, Washington, DC.

Table 3.3.2-1 (page 11 of 11)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	
14. Block Turbine Bypass and Cooldown Valves					
a. Manual Initiation	1,2,3	Trains A and D	F	SR 3.3.2.5	
b. Actuation Logic and Actuation Outputs	1,2,3	Trains A and D	S,T	SR 3.3.2.2 SR 3.3.2.3	
c. Low-Low T_{avg} Signal	$1,2,3^{(g)}$	3	M,N	SR 3.3.2.1 SR 3.3.2.2 SR 3.3.2.6 SR 3.3.2.7	
15. Manual Control of ESF Components					
a. Safety VDU	1, 2, 3, 4, 5, 6[(i)]	4 trains	O	SR 3.3.2.2 SR 3.3.2.9	DCD_09.01.02-53
b. COM-2	1, 2, 3, 4, 5, 6[(i)]	4 trains	P	SR 3.3.2.2	DCD_09.01.02-53
c. Actuation Logic and Actuation Outputs	Refer to LCO 3.4 through 3.7 for all requirements applicable to the controlled ESF components.			SR 3.3.2.2 SR 3.3.2.3	
16. Main Steam Relief Line Isolation					
a. Manual Initiation	1,2,3	2 trains per SG	F	SR 3.3.2.5	
b. Actuation Logic and Actuation Outputs	1,2,3	2 trains per SG	G	SR 3.3.2.2 SR 3.3.2.3	
c. Low Main Steam Line Pressure	$1,2,3^{(h)}$	3 per SG	M,N	SR 3.3.2.1 SR 3.3.2.2 SR 3.3.2.6 SR 3.3.2.7	

(g) Low-Low T_{avg} Signal for Cooldown Turbine Bypass Valves is required to be OPERABLE in MODE 3 above the setpoint of Low-Low T_{avg}. Low-Low T_{avg} Signal for Turbine Bypass Valves (except Cooldown Turbine Bypass Valves) is required to be OPERABLE in MODE 3.

(h) Except while manual cooling operation with MSRV or MSDV by the operator.

(i) Also applicable when one or more fuel assemblies are seated in the containment racks. DCD_09.01.02-53

3.8 ELECTRICAL POWER SYSTEMS

3.8.2 AC Sources - Shutdown

LCO 3.8.2 The following ac electrical power sources shall be OPERABLE:

a. One qualified circuit between the offsite transmission network and the onsite Class 1E ac electrical power distribution subsystem(s) required by LCO 3.8.10, "Distribution Systems - Shutdown" and

b. Two Class 1E Gas Turbine Generators (GTGs) capable of supplying two trains of the onsite Class 1E ac electrical power distribution subsystem(s) required by LCO 3.8.10.

APPLICABILITY: MODES 5 and 6,
During movement of irradiated fuel assemblies and when one or more irradiated fuel assemblies are seated in the containment racks. DCD_09.01.02-53

ACTIONS

---NOTE---
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required offsite circuit inoperable.	-------------------NOTE------------------- Enter applicable Conditions and Required Actions of LCO 3.8.10, with one required train de-energized as a result of Condition A. ---	
	A.1 Declare affected required feature(s) with no offsite power available inoperable.	Immediately
	OR	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	AND	

3.8 ELECTRICAL POWER SYSTEMS

3.8.5 DC Sources - Shutdown

LCO 3.8.5 DC electrical power subsystems shall be OPERABLE to support the dc electrical power distribution subsystems required by LCO 3.8.10, "Distribution Systems - Shutdown".

APPLICABILITY: MODES 5 and 6,
During movement of irradiated fuel assemblies and when one or more irradiated fuel assemblies are seated in the containment racks. DCD_09.01.02-53

ACTIONS

---NOTE---

LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One battery charger on one required train inoperable. AND The required redundant train(s) battery and charger OPERABLE.	A.1 Restore battery terminal voltage to greater than or equal to the minimum established float voltage.	2 hours
	AND	
	A.2 Verify battery float current ≤ [5] amps.	Once per 24 hours
	AND	
	A.3 Restore battery charger to OPERABLE status.	7 days

3.8 ELECTRICAL POWER SYSTEMS

3.8.8 Inverters - Shutdown

LCO 3.8.8 Inverters shall be OPERABLE to support the onsite Class 1E ac vital bus electrical power distribution subsystems required by LCO 3.8.10, "Distribution Systems - Shutdown."

APPLICABILITY: MODES 5 and 6,
During movement of irradiated fuel assemblies and when one or more irradiated fuel assemblies are seated in the containment racks. DCD_09.01.02-53

ACTIONS

---NOTE---
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required inverters inoperable.	A.1 Declare affected required feature(s) inoperable.	Immediately
	OR	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2.2 Suspend movement of irradiated fuel assemblies.	Immediately
	AND	
	A.2.3 Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration.	Immediately
	AND	
	A.2.4 Initiate action to restore required inverters to OPERABLE status.	Immediately

3.8 ELECTRICAL POWER SYSTEMS

3.8.10 Distribution Systems - Shutdown

LCO 3.8.10 The necessary portions of ac, dc, and ac vital bus electrical power distribution subsystems shall be OPERABLE to support equipment required to be OPERABLE.

APPLICABILITY: MODES 5 and 6,
During movement of irradiated fuel assemblies and when one or more irradiated fuel assemblies are seated in the containment racks. DCD_09.01.02-53

ACTIONS

--NOTE--

LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required ac, dc, or ac vital bus electrical power distribution subsystems inoperable.	A.1 Declare associated supported required feature(s) inoperable.	Immediately
	OR	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2.2 Suspend movement of irradiated fuel assemblies.	Immediately
	AND	
	A.2.3 Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration.	Immediately
	AND	

3.9 REFUELING OPERATIONS

3.9.1 Boron Concentration

LCO 3.9.1 Boron concentrations of the Reactor Coolant System, the refueling canal, and the refueling cavity shall be maintained within the limit specified in the COLR.

APPLICABILITY: MODE 6 and when one or more fuel assemblies are seated in the containment racks. DCD_09.01.02-53

--NOTE--
Only applicable to the refueling canal and refueling cavity when connected to the RCS.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Boron concentration not within limit.	A.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2 Suspend positive reactivity additions.	Immediately
	AND	
	A.3 Initiate action to restore boron concentration to within limit.	Immediately

3.9 REFUELING OPERATIONS

3.9.2 Unborated Water Source Isolation Valves

LCO 3.9.2 Each valve used to isolate unborated water sources shall be secured in the closed position.

APPLICABILITY: MODE 6 and when one or more fuel assemblies are seated in the containment racks. DCD_09.01.02-53

ACTIONS

---NOTE---
Separate Condition entry is allowed for each unborated water source isolation valve.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ------------NOTE---------- Required Action A.3 must be completed whenever Condition A is entered. ------------------------------ One or more valves not secured in closed position.	A.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2 Initiate actions to secure valve in closed position.	Immediately
	AND	
	A.3 Perform SR 3.9.1.1.	4 hours

3.9 REFUELING OPERATIONS

3.9.5 Residual Heat Removal (RHR) and Coolant Circulation - High Water Level

LCO 3.9.5 Two RHR loops shall be OPERABLE and in operation.

---NOTE---
The required RHR loops may be removed from operation for ≤ 1 hour per 8 hour period, provided no operations are permitted that would cause introduction of coolant into the Reactor Coolant System with boron concentration less than that required to meet the minimum required boron concentration of LCO 3.9.1.

APPLICABILITY: MODE 6 with the water level ≥ 23 ft above the top of reactor vessel flange and when one or more irradiated fuel assemblies are seated in the containment racks. DCD_09.01.02-53

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RHR loop requirements not met.	A.1 Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet the boron concentration of LCO 3.9.1.	Immediately
	AND	
	A.2 Suspend loading irradiated fuel assemblies in the core.	Immediately
	AND	
	A.3 Initiate action to satisfy RHR loop requirements.	Immediately
	AND	

3.9 REFUELING OPERATIONS

3.9.7 Refueling Cavity Water Level

LCO 3.9.7 Refueling cavity water level shall be maintained ≥ 23 ft above the top of reactor vessel flange.

APPLICABILITY: During movement of irradiated fuel assemblies within containment and when one or more irradiated fuel assemblies are seated in the containment racks. DCD_09.01.02-53

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Refueling cavity water level not within limit.	A.1 Suspend movement of irradiated fuel assemblies within containment. AND A.2 Initiate actions to restore refueling cavity water level to within limits.	Immediately Immediately

DCD_09.01.02-53

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.7.1	Verify refueling cavity water level is ≥ 23 ft above the top of reactor vessel flange.	[24 hours OR In accordance with the Surveillance Frequency Control Program]

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, and

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:

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

b. $k_{eff} \leq 0.95$ during normal conditions if fully flooded with unborated water, which includes an allowance for uncertainties as described in Subsection 9.1.1 of the DCD, and DCD_09.01.02-51

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

4.3.2 Drainage

BASES

LCO 3.0.3 (continued)

allowable limit of 13 hours. Therefore, if remedial measures are completed that would permit a return to MODE 1, a penalty is not incurred by having to reach a lower MODE of operation in less than the total time allowed.

In MODES 1, 2, 3, and 4, LCO 3.0.3 provides actions for Conditions not covered in other Specifications. The requirements of LCO 3.0.3 do not apply in MODES 5 and 6 and when one or more fuel assemblies are seated in the containment racks because the unit is already in the most restrictive Condition required by LCO 3.0.3. The requirements of LCO 3.0.3 do not apply in other specified conditions of the Applicability (unless in MODE 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken. DCD_09.01.02-53

Exceptions to LCO 3.0.3 are provided in instances where requiring a unit shutdown, in accordance with LCO 3.0.3, would not provide appropriate remedial measures for the associated condition of the unit. An example of this is in LCO 3.7.12, "Spent Fuel pit Water Level." LCO 3.7.12 has an Applicability of "During movement of irradiated fuel assemblies in the spent fuel pit." Therefore, this LCO can be applicable in any or all MODES. If the LCO and the Required Actions of LCO 3.7.12 are not met while in MODE 1, 2, or 3, there is no safety benefit to be gained by placing the unit in a shutdown condition. The Required Action of LCO 3.7.12 of "Suspend movement of irradiated fuel assemblies in the spent fuel pit" is the appropriate Required Action to complete in lieu of the actions of LCO 3.0.3. These exceptions are addressed in the individual Specifications.

LCO 3.0.4

LCO 3.0.4 establishes limitations on changes in MODES or other specified conditions in the Applicability when an LCO is not met. It allows placing the unit in a MODE or other specified condition stated in that Applicability (e.g., the Applicability desired to be entered) when unit conditions are such that the requirements of the LCO would not be met, in accordance with LCO 3.0.4.a, LCO 3.0.4.b, or LCO 3.0.4.c.

LCO 3.0.4.a allows entry into a MODE or other specified condition in the Applicability with the LCO not met when the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time. Compliance with Required Actions that permit continued operation of the unit for an unlimited period of time in a MODE or other specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of the unit before or after the MODE change. Therefore, in such cases, entry into a MODE or other specified condition in the Applicability may be made in accordance with the provisions of the Required Actions.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Control of ESF Components is required for some ESF systems in all MODES, S-VDU must be OPERABLE to support ESF components in MODES 1, 2, 3, 4, 5 and 6, and when one or more fuel assemblies are seated in the containment racks. DCD_09.01.02-53

b. Manual Control of ESF Components - COM-2

COM-2 combines the manual control signals from non-safety Operational VDUs with the manual control signals from S-VDUs. The combined signal is interfaced to the Actuation Logic in the SLS for manual control of ESF components. Since COM-2 controls ESF components assigned to all four trains as explained above for the S-VDU, some of which are required in all MODES, four COM-2 trains are required in MODES 1, 2, 3, 4, 5 and 6, and when one or more fuel assemblies are seated in the containment racks. DCD_09.01.02-53

c. Manual Control of ESF Components - Actuation Logic and Actuation Outputs

The Actuation Logic and Actuation Outputs for the Manual Control of ESF Components Function is implemented in the SLS. For ESFAS components, the SLS combines the automatic actuation signals from the ESFAS with the manual control signals from COM-2. For all ESF components the SLS generates Actuation Outputs, based on automatic and/or manual control signals, which control the state of the ESF components. For the Manual Control of ESF Components Function, the Actuation Logic and Actuation Outputs within any SLS controller must be OPERABLE in the same MODES and for the same trains as for the required ESF components. LCO 3.4 through 3.7 provide MODE and train requirements applicable to ESF components.

16. Main Steam Relief Line Isolation

The Main Steam Relief Line Isolation is to prevent the overcooling of the reactor coolant system in the event of MSRV malfunction. For this objective, the Main Steam Relief Line Isolation is automatically actuated by the Low Main Steam Line Pressure signal. The Function may also be actuated manually. The Main Steam Relief Line Isolation function is actuated separately for each steam relief line, either manually or automatically.

The control of the MSRV and the MSRVBV are distributed to two different trains.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.2 AC Sources - Shutdown

BASES

BACKGROUND

A description of the ac sources is provided in the Bases for LCO 3.8.1, "AC Sources - Operating."

APPLICABLE SAFETY ANALYSES

The OPERABILITY of the minimum ac sources during MODES 5 and 6, ~~and~~ during movement of irradiated fuel assemblies and when one or more irradiated fuel assemblies are seated in the containment racks ensures that: DCD_09.01.02-53

a. The unit can be maintained in the shutdown or refueling condition for extended periods,

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status, and

c. Adequate ac electrical power is provided to mitigate events postulated during shutdown.

In general, when the unit is shut down, the Technical Specifications requirements ensure that the unit has the capability to mitigate the consequences of postulated accidents. However, assuming a single failure and concurrent loss of all offsite or all onsite power is not required. The rationale for this is based on the fact that many Postulated Accident (PA) that are analyzed in MODES 1, 2, 3, and 4 have no specific analyses in MODES 5 and 6. Worst case bounding events are deemed not credible in MODES 5 and 6 because the energy contained within the reactor pressure boundary, reactor coolant temperature and pressure, and the corresponding stresses result in the probabilities of occurrence being significantly reduced or eliminated, and in minimal consequences. These deviations from PA analysis assumptions and design requirements during shutdown conditions are allowed by the LCO for required systems.

During MODES 1, 2, 3, and 4, various deviations from the analysis assumptions and design requirements are allowed within the Required Actions. This allowance is in recognition that certain testing and maintenance activities must be conducted provided an acceptable level of risk is not exceeded. During MODES 5 and 6, performance of a significant number of required testing and maintenance activities is also required. In MODES 5 and 6, the activities are generally planned and administratively controlled. Relaxations from MODE 1, 2, 3, and 4 LCO requirements are acceptable during shutdown modes based on:

a. The fact that time in an outage is limited. This is a risk prudent goal as well as a utility economic consideration.

BASES

LCO (continued)

The Class 1E GTGs must be capable of starting, accelerating to rated speed and voltage, and connecting to its respective Class 1E on detection of bus undervoltage. This sequence must be accomplished within 100 seconds. The Class 1E GTGs must be capable of accepting required loads within the assumed loading sequence intervals, and continue to operate until offsite power can be restored to the Class 1E buses. These capabilities are required to be met from a variety of initial conditions such as Class 1E GTG in standby with the engine hot and Class 1E GTG in standby at ambient conditions.

Proper sequencing of loads, including tripping of nonessential loads, is a required function for Class 1E GTG OPERABILITY.

In addition, proper sequencer operation is an integral part of offsite circuit OPERABILITY since its inoperability impacts on the ability to start and maintain energized loads required OPERABLE by LCO 3.8.10.

APPLICABILITY

The ac sources required to be OPERABLE in MODES 5 and 6, ~~and~~ during movement of irradiated fuel assemblies and when one or more irradiated fuel assemblies are seated in the containment racks provide assurance that: DCD_09.01.02-53

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel assemblies in the core,

b. Systems needed to mitigate a fuel handling accident are available,

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available, and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

The ac power requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.1.

ACTIONS

LCO 3.0.3 is not applicable while in MODE 5 or 6 or when one or more irradiated fuel assemblies are seated in the containment racks. However, since irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Entering LCO 3.0.3, while in MODE 1, 2, 3, or 4 would require the unit to be shutdown unnecessarily. DCD_09.01.02-53

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.5 DC Sources - Shutdown

BASES

BACKGROUND	A description of the dc sources is provided in the Bases for LCO 3.8.4, "DC Sources - Operating."

APPLICABLE SAFETY ANALYSES

The initial conditions of Anticipated Operational Occurrence (AOO) and Postulated Accident (PA) analyses in Chapter 6 (Ref. 1) and Chapter 15 (Ref. 2), assume that Engineered Safety Feature systems are OPERABLE. The dc electrical power system provides normal and emergency dc electrical power for the Class 1E Gas Turbine Generators (GTGs), emergency auxiliaries, and control and switching during all MODES of operation.

The OPERABILITY of the dc subsystems is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

The OPERABILITY of the minimum dc electrical power sources during MODES 5 and 6, ~~and~~ during movement of irradiated fuel assemblies and when one or more irradiated fuel assemblies are seated in the containment racks ensures that: DCD_09.01.02-53

a. The unit can be maintained in the shutdown or refueling condition for extended periods,

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status, and

c. Adequate dc electrical power is provided to mitigate events postulated during shutdown.

In general, when the unit is shut down, the Technical Specifications requirements ensure that the unit has the capability to mitigate the consequences of postulated accidents. However, assuming a single failure and concurrent loss of all offsite or all onsite power is not required. The rationale for this is based on the fact that many PA that are analyzed in MODES 1, 2, 3, and 4 have no specific analyses in MODES 5 and 6 because the energy contained within the reactor pressure boundary, reactor coolant temperature and pressure, and the corresponding stresses result in the probabilities of occurrence being significantly reduced or eliminated, and in minimal consequences. These deviations from PA analysis assumptions and design requirements during shutdown conditions are allowed by the LCO for required systems.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The shutdown Technical Specification requirements are designed to ensure that the unit has the capability to mitigate the consequences of certain postulated accidents. Worst case PA which are analyzed for operating MODES are generally viewed not to be a significant concern during shutdown MODES due to the lower energies involved. The Technical Specifications therefore require a lesser complement of electrical equipment to be available during shutdown than is required during operating MODES. More recent work completed on the potential risks associated with shutdown, however, have found significant risk associated with certain shutdown evolutions. As a result, in addition to the requirements established in the Technical Specifications, the industry has adopted NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management," as an Industry initiative to manage shutdown tasks and associated electrical support to maintain risk at an acceptable low level. This may require the availability of additional equipment beyond that required by the shutdown Technical Specifications.

The dc sources satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The dc electrical power is made up of subsystems. Each subsystem consists of one battery, one battery charger per battery, and the corresponding control equipment and interconnecting cabling within the train. A subsystem is required to be OPERABLE to support the associated required trains of the distribution systems specified by LCO 3.8.10, "Distribution Systems - Shutdown." This ensures the availability of sufficient dc electrical power sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

APPLICABILITY

The dc electrical power sources required to be OPERABLE in MODES 5 and 6, ~~and~~ during movement of irradiated fuel assemblies and when one or more irradiated fuel assemblies are seated in the containment racks~~,~~ provide assurance that: DCD_09.01.02-53

a. Required features to provide adequate coolant inventory makeup are available for the irradiated fuel assemblies in the core,

b. Required features needed to mitigate a fuel handling accident are available,

c. Required features necessary to mitigate the effects of events that can lead to core damage during shutdown are available, and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

BASES

APPLICABILITY (continued)

The dc electrical power requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.4, “DC Sources – Operating”.

ACTIONS

LCO 3.0.3 is not applicable while in MODE 5 or 6 or when one or more irradiated fuel assemblies are seated in the containment racks. However, since irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Entering LCO 3.0.3, while in MODE 1, 2, 3, or 4 would require the unit to be shutdown unnecessarily. DCD_09.01.02-53

A.1, A.2, and A.3

Condition A represents one required train with one battery charger inoperable (e.g., the voltage limit of SR 3.8.4.1 is not maintained). The ACTIONS provide a tiered response that focuses on returning the battery to the fully charged state and restoring a fully qualified charger to OPERABLE status in a reasonable time period. Required Action A.1 requires that the battery terminal voltage be restored to greater than or equal to the minimum established float voltage within 2 hours. This time provides for returning the inoperable charger to OPERABLE status or providing an alternate means of restoring battery terminal voltage to greater than or equal to the minimum established float voltage. Restoring the battery terminal voltage to greater than or equal to the minimum established float voltage provides good assurance that, within 24 hours, the battery will be restored to its fully charged condition (Required Action A.2) from any discharge that might have occurred due to the charger inoperability.

A discharged battery having terminal voltage of at least the minimum established float voltage indicates that the battery is on the exponential charging current portion (the second part) of its recharge cycle. The time to return a battery to its fully charged state under this condition is simply a function of the amount of the previous discharge and the recharge characteristic of the battery. Thus there is good assurance of fully recharging the battery within 24 hours, avoiding a premature shutdown with its own attendant risk.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.8 Inverters - Shutdown

BASES

BACKGROUND	A description of the inverters is provided in the Bases for LCO 3.8.7, "Inverters - Operating."

APPLICABLE SAFETY ANALYSES

The initial conditions of Anticipated Operational Occurrence (AOO) and Postulated Accident (PA) analyses in Chapter 6 (Ref. 1) and Chapter 15 (Ref. 2), assume Engineered Safety Feature systems are OPERABLE. The dc to ac inverters are designed to provide the required capacity, capability, redundancy, and reliability to ensure the availability of necessary power to the Reactor Protective System and Engineered Safety Features Actuation System instrumentation and controls so that the fuel, Reactor Coolant System, and containment design limits are not exceeded.

The OPERABILITY of the inverters is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

The OPERABILITY of the minimum inverters to each ac vital bus during MODES 5 and 6, during movement of irradiated fuel assemblies and when one or more irradiated fuel assemblies are seated in the containment racks ensures that: DCD_09.01.02-53

a. The unit can be maintained in the shutdown or refueling condition for extended periods,

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status, and

c. Adequate power is available to mitigate events postulated during shutdown. DCD_09.01.02-53

In general, when the unit is shut down, the Technical Specifications requirements ensure that the unit has the capability to mitigate the consequences of postulated accidents. However, assuming a single failure and concurrent loss of all offsite or all onsite power is not required. The rationale for this is based on the fact that many PA that are analyzed in MODES 1, 2, 3, and 4 have no specific analyses in MODES 5 and 6 because the energy contained within the reactor pressure boundary, reactor coolant temperature and pressure, and the corresponding stresses result in the probabilities of occurrence being significantly reduced or eliminated, and in minimal consequences. These deviations from PA analysis assumptions and design requirements during shutdown conditions are allowed by the LCO for required systems.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The shutdown Technical Specification requirements are designed to ensure that the unit has the capability to mitigate the consequences of certain postulated accidents. Worst case PA which are analyzed for operating MODES are generally viewed not to be a significant concern during shutdown MODES due to the lower energies involved. The Technical Specifications therefore require a lesser complement of electrical equipment to be available during shutdown than is required during operating MODES. More recent work completed on the potential risks associated with shutdown, however, have found significant risk associated with certain shutdown evolutions. As a result, in addition to the requirements established in the Technical Specifications, the industry has adopted NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management," as an Industry initiative to manage shutdown tasks and associated electrical support to maintain risk at an acceptable low level. This may require the availability of additional equipment beyond that required by the shutdown Technical Specifications.

The inverters were previously identified as part of the distribution system and, as such, satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The inverters ensure the availability of electrical power for the instrumentation for systems required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence or a postulated DBA. The battery powered inverters provide uninterruptible supply of ac electrical power to the ac vital buses even if the normal power from the 480Vac safety buses are de-energized. OPERABILITY of the inverters requires that the ac vital bus be powered by the inverter. This ensures the availability of sufficient inverter power sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown).

APPLICABILITY

The inverters required to be OPERABLE in MODES 5 and 6, ~~and~~ during movement of irradiated fuel assemblies and when one or more irradiated fuel assemblies are seated in the containment racks provide assurance that: DCD_09.01.02-53

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core,

b. Systems needed to mitigate a fuel handling accident are available,

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available, and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

BASES

APPLICABILITY (continued)

Inverter requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.7, “Inverters - Operating.”

ACTIONS

LCO 3.0.3 is not applicable while in MODE 5 or 6 or when one or more irradiated fuel assemblies are seated in the containment racks. However, since irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Entering LCO 3.0.3, while in MODE 1, 2, 3, or 4 would require the unit to be shutdown unnecessarily. DCD_09.01.02-53

A.1, A.2.1, A.2.2, A.2.3, and A.2.4

If two trains are required by LCO 3.8.10, "Distribution Systems - Shutdown," the remaining OPERABLE Inverters may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS, irradiated fuel movement, and operations with a potential for positive reactivity additions. By the allowance of the option to declare required features inoperable with the associated inverter(s) inoperable, appropriate restrictions will be implemented in accordance with the affected required features LCOs' Required Actions. In many instances, this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies, and operations involving positive reactivity additions) that could result in loss of required SDM (MODE 5) or boron concentration (MODE 6). Suspending positive reactivity additions that could result in failure to meet the minimum SDM or boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that what would be required in the RCS for minimum SDM or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes including temperature increases when operating with a positive MTC must also be evaluated to ensure they do not result in a loss of required SDM.

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required inverters and to continue this action until restoration is accomplished in order to provide the necessary inverter power to the unit safety systems.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.10 Distribution Systems - Shutdown

BASES

BACKGROUND

A description of the ac, dc, and ac vital bus electrical power distribution systems is provided in the Bases for LCO 3.8.9, "Distribution Systems - Operating."

APPLICABLE SAFETY ANALYSES

The initial conditions of Anticipated Operational Occurrence (AOO) and Postulated Accident (PA) analyses in Chapter 6 (Ref. 1) and Chapter 15 (Ref. 2), assume Engineered Safety Feature (ESF) systems are OPERABLE. The ac, dc, and ac vital bus electrical power distribution systems are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System, and containment design limits are not exceeded.

The OPERABILITY of the ac, dc, and ac vital bus electrical power distribution system is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

The OPERABILITY of the minimum ac, dc, and ac vital bus electrical power distribution subsystems during MODES 5 and 6, ~~and~~ during movement of irradiated fuel assemblies and when one or more irradiated fuel assemblies are seated in the containment racks ensures that: DCD_09.01.02-53

a. The unit can be maintained in the shutdown or refueling condition for extended periods,

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status, and

c. Adequate power is provided to mitigate events postulated during shutdown, such as a fuel handling accident.

The ac and dc electrical power distribution systems satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Various combinations of subsystems, equipment, and components are required OPERABLE by other LCOs, depending on the specific plant condition. Implicit in those requirements is the required OPERABILITY of necessary support required features. This LCO explicitly requires energization of the portions of the electrical distribution system necessary to support OPERABILITY of required systems, equipment, and components - all specifically addressed in each LCO and implicitly required via the definition of OPERABILITY.

LCO (continued)

Maintaining these portions of the distribution system energized ensures the availability of sufficient power to operate the unit in a safe manner to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents). The necessary portions in each mode are controlled by administrative control.

APPLICABILITY

The ac and dc electrical power distribution subsystems required to be OPERABLE in MODES 5 and 6, ~~and~~ during movement of irradiated fuel assemblies and when one or more irradiated fuel assemblies are seated in the containment racks, provide assurance that: DCD_09.01.02-53

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core,

b. Systems needed to mitigate a fuel handling accident are available,

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available, and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition and refueling condition.

The ac, dc, and ac vital bus electrical power distribution subsystems requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.9, “Distribution Systems - Operating.”

ACTIONS

LCO 3.0.3 is not applicable while in MODE 5 or 6 or when one or more irradiated fuel assemblies are seated in the containment racks. However, since irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Entering LCO 3.0.3, while in MODE 1, 2, 3, or 4 would require the unit to be shutdown unnecessarily. DCD_09.01.02-53

B 3.9 REFUELING OPERATIONS

B 3.9.1 Boron Concentration

BASES

BACKGROUND

The limit on the boron concentrations of the Reactor Coolant System (RCS), the refueling canal, and the refueling cavity during refueling ensures that the reactor remains subcritical during MODE 6 and when one or more fuel assemblies are seated in the containment racks. Refueling boron concentration is the soluble boron concentration in the coolant in each of these volumes having direct access to the reactor core during refueling. DCD_09.01.02-53

The soluble boron concentration offsets the core reactivity and is measured by chemical analysis of a representative sample of the coolant in each of the volumes. The refueling boron concentration limit is specified in the COLR. Plant procedures ensure the specified boron concentration in order to maintain an overall core reactivity of $k_{eff} \leq 0.95$ during fuel handling, with control rods and fuel assemblies assumed to be in the most adverse configuration (least negative reactivity) allowed by plant procedures.

GDC 26 of 10 CFR 50, Appendix A, requires that two independent reactivity control systems of different design principles be provided (Ref. 1). One of these systems must be capable of holding the reactor core subcritical under cold conditions. The Chemical and Volume Control System (CVCS) is the system capable of maintaining the reactor subcritical in cold conditions by maintaining the boron concentration.

The reactor is brought to shutdown conditions before beginning operations to open the reactor vessel for refueling. After the RCS is cooled and depressurized and the vessel head is unbolted, the head is slowly removed to form the refueling cavity. The refueling canal and the refueling cavity are then flooded with borated water from the refueling water storage pit by the use of the Containment Spray (CS)/Residual Heat Removal (RHR) System pumps. The refueling canal is flooded from the Refueling Water Auxiliary Tank by using the Refueling Water Recirculation Pumps.

The pumping action of the RHR System in the RCS and the natural circulation due to thermal driving heads in the reactor vessel and refueling cavity mix the added concentrated boric acid with the water in the refueling canal. The RHR System is in operation during refueling (see LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level," and LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level") to provide forced circulation in the RCS and assist in maintaining the boron concentrations in the RCS, the refueling canal, and the refueling cavity above the COLR limit.

BASES

APPLICABLE SAFETY ANALYSES

During refueling operations, the reactivity condition of the core is consistent with the initial conditions assumed for the boron dilution accident in the accident analysis and is conservative for MODE 6 and when one or more fuel assemblies are seated in the containment racks. The boron concentration limit specified in the COLR is based on the core reactivity at the beginning of each fuel cycle (the end of refueling) and includes an uncertainty allowance. DCD_09.01.02-53

The required boron concentration and the plant refueling procedures that verify the correct fuel loading plan (including full core mapping) ensure that the k_{eff} of the core will remain ≤ 0.95 during the refueling operation. Hence, at least a 5% $\Delta k/k$ margin of safety is established during refueling.

During refueling, the water volume in the spent fuel pit, the transfer canal, the refueling canal, the refueling cavity, and the reactor vessel form a single mass. As a result, the soluble boron concentration is relatively the same in each of these volumes.

The limiting boron dilution accident analyzed occurs in MODE 5 (Ref. 2). A detailed discussion of this event is provided in Bases B 3.1.1, "SHUTDOWN MARGIN (SDM)."

The RCS boron concentration satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requires that a minimum boron concentration be maintained in the RCS, the refueling canal, and the refueling cavity while in MODE 6 and when one or more fuel assemblies are seated in the containment racks. The boron concentration limit specified in the COLR ensures that a core k_{eff} of ≤ 0.95 is maintained during fuel handling operations. Violation of the LCO could lead to an inadvertent criticality during MODE 6. DCD_09.01.02-53

APPLICABILITY

This LCO is applicable in MODE 6 to ensure that the fuel in the reactor vessel will remain subcritical. This LCO is also applicable when one or more fuel assemblies are seated in the containment racks to preserve the assumptions used in potential criticality accident scenarios. The required boron concentration ensures a k_{eff} ≤ 0.95. Above MODE 6, LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," ensures that an adequate amount of negative reactivity is available to shut down the reactor and maintain it subcritical. DCD_09.01.02-53

The Applicability is modified by a Note. The Note states that the limits on boron concentration are only applicable to the refueling canal and the refueling cavity when those volumes are connected to the RCS. When the refueling canal and the refueling cavity are isolated from the RCS, no potential path for boron dilution exists.

B 3.9 REFUELING OPERATIONS

B 3.9.2 Unborated Water Source Isolation Valves

BASES

BACKGROUND	During MODE 6 operations and when one or more fuel assemblies are seated in the containment racks, all isolation valves for reactor makeup water sources containing unborated water that are connected to the Reactor Coolant System (RCS) must be closed to prevent unplanned boron dilution of the reactor coolant. The isolation valves must be secured in the closed position. The Chemical and Volume Control System is capable of supplying borated and unborated water to the RCS through various flow paths. Since a positive reactivity addition made by reducing the boron concentration is inappropriate during MODE 6 and when one or more fuel assemblies are seated in the containment racks, isolation of all unborated water sources prevents an unplanned boron dilution.	DCD_09.01.02-53 DCD_09.01.02-53
APPLICABLE SAFETY ANALYSES	The possibility of an inadvertent boron dilution event (Ref. 1) occurring during MODE 6 refueling operations and when one or more fuel assemblies are seated in the containment racks is precluded by adherence to this LCO, which requires that potential dilution sources be isolated. Closing the required valves during refueling operations prevents the flow of unborated water to the filled portion of the RCS. The valves are used to isolate unborated water sources. These valves have the potential to indirectly allow dilution of the RCS boron concentration in MODE 6 and when one or more fuel assemblies are seated in the containment racks. By isolating unborated water sources, a safety analysis for an uncontrolled boron dilution accident in accordance with the Standard Review Plan (Ref. 2) is not required for MODE 6 and when one or more fuel assemblies are seated in the containment racks. The RCS boron concentration satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).	DCD_09.01.02-53 DCD_09.01.02-53 DCD_09.01.02-53
LCO	This LCO requires that flow paths to the RCS from unborated water sources be isolated to prevent unplanned boron dilution during MODE 6 and when one or more fuel assemblies are seated in the containment racks and thus avoid a reduction in SDM.	DCD_09.01.02-53
APPLICABILITY	In MODE 6 and when one or more fuel assemblies are seated in the containment racks, this LCO is applicable to prevent an inadvertent boron dilution event by ensuring isolation of all sources of unborated water to the RCS. For all other MODES, the boron dilution accident was analyzed and was found to be capable of being mitigated.	DCD_09.01.02-53

SURVEILLANCE REQUIREMENTS

SR 3.9.2.1

These valves are to be secured closed to isolate possible dilution paths. The likelihood of a significant reduction in the boron concentration during MODE 6 operations and when one or more fuel assemblies are seated in the containment racks is remote due to the large mass of borated water in the refueling cavity and the fact that all unborated water sources are isolated, precluding a dilution. The boron concentration is checked every 72 hours during MODE 6 and when one or more fuel assemblies are seated in the containment racks under SR 3.9.1.1. This Surveillance demonstrates that the valves are closed through a system walkdown. [The 31 day Frequency is based on engineering judgment and is considered reasonable in view of other administrative controls that will ensure that the valve opening is an unlikely possibility. OR The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.] DCD_09.01.02-53

REFERENCES

1. Subsection 15.4.6.

2. NUREG-0800, Section 15.4.6.

B 3.9 REFUELING OPERATIONS

B 3.9.5 Residual Heat Removal (RHR) and Coolant Circulation - High Water Level

BASES

BACKGROUND

The purpose of the RHR System in MODE 6 and when one or more irradiated fuel assemblies are seated in the containment racks is to remove decay heat and sensible heat from the Reactor Coolant System (RCS) and refueling cavity water, as required by GDC 34, to provide mixing of borated coolant and to prevent boron stratification (Ref. 1). Heat is removed from the RCS by circulating reactor coolant through the Containment Spray (CS)/RHR heat exchanger(s), where the heat is transferred to the Component Cooling Water System. The coolant is then returned to the RCS via the RCS cold leg(s). Operation of the RHR System for normal cooldown or decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by controlling the flow of reactor coolant through the CS/RHR heat exchanger(s) and the bypass line(s). Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the RHR System. DCD_09.01.02-53

The Safeguards Component Area HVAC System is a support system that provides temperature control for the CS/RHR Pump Room and CS/RHR Heat Exchanger Room, and includes electric heating coils, cooling coils, fans, ductwork, dampers, and instrumentation and controls necessary to perform the support function. The Essential Chilled Water System is a support system and provides chilled water to the air handling unit cooling coil. The Essential Service Water System supports operation of the essential chiller. For each RHR loop required to be OPERABLE, the associated train of Safeguards Component Area HVAC System, including its associated train of the Essential Chilled Water System and Essential Service Water System, must be in operation, or available to operate on demand, and capable of performing its related support function.

BASES

APPLICABLE SAFETY ANALYSES

While there is no explicit analysis assumptions for the decay heat removal function of the RHR System in MODE 6 or when one or more irradiated fuel assemblies are seated in the containment racks, if the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to a loss of refueling cavity water level. Additionally, boiling of the reactor coolant could lead to a reduction in boron concentration in the coolant due to boron plating out on components near the areas of the boiling activity. The loss of reactor coolant and the reduction of boron concentration in the reactor coolant would eventually challenge the integrity of the fuel cladding, which is a fission product barrier. Two trains of the RHR System are required to be OPERABLE and at least one train of RHR System is operating in MODE 6 or when one or more irradiated fuel assemblies are seated in the containment racks, with the water level ≥ 23 ft above the top of the reactor vessel flange, to prevent this challenge. The LCO does permit the CS/RHR pumps to be removed from operation for short durations, under the condition that the boron concentration is not reduced. This conditional stopping of the CS/RHR pumps does not result in a challenge to the fission product barrier. DCD_09.01.02-53 DCD_09.01.02-53

RHR and Coolant Circulation – High Water Level satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

LCO

Only two RHR loops are required for decay heat removal in MODE 6 and when one or more irradiated fuel assemblies are seated in the containment racks, with the water level ≥ 23 ft above the top of the reactor vessel flange. Only two RHR loops are required to be OPERABLE, because the volume of water above the reactor vessel flange provides backup decay heat removal capability. At least two RHR loops must be OPERABLE and in operation to provide: DCD_09.01.02-53

a. Removal of decay heat,

b. Mixing of borated coolant to minimize the possibility of criticality, and

c. Indication of reactor coolant temperature.

LCO (continued)

An OPERABLE RHR loop includes a CS/RHR pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path and to determine the low end temperature. The flow path starts in one of the RCS hot legs and is returned to the RCS cold legs. Management of gas voids is important to RHR System OPERABILITY.

The LCO is modified by a Note that allows the required operating RHR loops to be removed from operation for up to 1 hour per 8 hour period, provided no operations are permitted that would dilute the RCS boron concentration by introduction of coolant into the RCS with boron concentration less than required to meet the minimum boron concentration of LCO 3.9.1. Boron concentration reduction with coolant at boron concentrations less than required to assure the RCS boron concentration is maintained is prohibited because uniform concentration distribution cannot be ensured without forced circulation. This permits operations such as core mapping or alterations in the vicinity of the reactor vessel hot leg nozzles and RCS to RHR isolation valve testing. During this 1 hour period, decay heat is removed by natural convection to the large mass of water in the refueling cavity.

APPLICABILITY

Two RHR loops must be OPERABLE and in operation in MODE 6 and when one or more irradiated fuel assemblies are seated in the containment racks, with the water level ≥ 23 ft above the top of the reactor vessel flange, to provide decay heat removal and mixing of the borated coolant. The 23 ft water level was selected because it corresponds to the 23 ft requirement established for fuel movement in LCO 3.9.7, "Refueling Cavity Water Level." Requirements for the RHR System in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS), and Section 3.5, Emergency Core Cooling Systems (ECCS). RHR loop requirements in MODE 6 with the water level < 23 ft are located in LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level." DCD_09.01.02-53

ACTIONS

RHR loop requirements are met by having two RHR loops OPERABLE and in operation, except as permitted in the Note to the LCO.

A.1

If RHR loop requirements are not met, there will be no forced circulation or insufficient forced circulation to provide mixing to establish uniform boron concentrations. Suspending positive reactivity additions that could result in failure to meet the minimum boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that what would be required in the RCS for minimum refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation.

A.2

If RHR loop requirements are not met, actions shall be taken immediately to suspend loading of irradiated fuel assemblies in the core. With no forced circulation cooling, decay heat removal from the core occurs by natural convection to the heat sink provided by the water above the core. A minimum refueling water level of 23 ft above the reactor vessel flange provides an adequate available heat sink. Suspending any operation that would increase decay heat load, such as loading a fuel assembly, is a prudent action under this condition.

A.3

If RHR loop requirements are not met, actions shall be initiated and continued in order to satisfy RHR loop requirements. With the unit in MODE 6 or when one or more irradiated fuel assemblies are seated in the containment racks and the refueling water level ≥ 23 ft above the top of the reactor vessel flange, corrective actions shall be initiated immediately. DCD_09.01.02-53

A.4, A.5, A.6.1, and A.6.2

If no RHR is in operation, the following actions must be taken:

a. The equipment hatch must be closed and secured with [four] bolts,

b. One door in each air lock must be closed, and

B 3.9 REFUELING OPERATIONS

B 3.9.7 Refueling Cavity Water Level

BASES

BACKGROUND

The movement of irradiated fuel assemblies within containment requires a minimum water level of 23 ft above the top of the reactor vessel flange. During refueling, this maintains sufficient water level in the containment, refueling canal, fuel transfer canal, refueling cavity, and spent fuel pit. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to within values reported in Chapter 15.

When one or more irradiated fuel assembles are seated in the containment racks, maintaining the specified water level in the refueling cavity provides shielding, minimizes the general area dose and provides an adequate available heat sink. DCD_09.01.02-53

APPLICABLE SAFETY ANALYSES

During movement of irradiated fuel assemblies, the water level in the refueling canal and the refueling cavity is an initial condition design parameter in the analysis of a fuel handling accident in containment, as postulated by Regulatory Guide 1.183 (Ref. 1). A minimum water level of 23 ft allows a decontamination factor of 200 (Appendix B2 of Ref. 1) to be used in the accident analysis for iodine. This relates to the assumption that 99.5% of the total iodine released from the pellet to cladding gap of all the dropped fuel assembly rods is retained by the refueling cavity water.

The fuel handling accident analysis inside containment is described in Reference 2. With a minimum water level of 23 ft and a minimum decay time of 24 hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water and offsite doses are maintained within allowable limits (Ref. 3).

Refueling cavity water level satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

A minimum refueling cavity water level of 23 ft above the reactor vessel flange is required to ensure that the radiological consequences of a postulated fuel handling accident inside containment are within acceptable limits, as calculated in Reference 2.

When one or more irradiated fuel assembles are seated in the containment racks, maintaining the specified water level in the refueling cavity provides shielding, minimizes the general area dose and provides an adequate available heat sink. DCD_09.01.02-53

APPLICABILITY

LCO 3.9.7 is applicable when moving irradiated fuel assemblies within containment and when one or more irradiated fuel assemblies are seated in the containment racks. The LCO minimizes the possibility of a fuel handling accident in containment that is beyond the assumptions of the safety analysis. If irradiated fuel assemblies are not being moved in containment, there can be no significant radioactivity release as a result of a postulated fuel handling accident. Requirements for fuel handling accidents in the spent fuel pit are covered by LCO 3.7.12, "Fuel Storage Pit Water Level." DCD_09.01.02-53

When one or more irradiated fuel assembles are seated in the containment racks, maintaining the specified water level in the refueling cavity provides shielding, minimizes the general area dose and provides an adequate available heat sink. DCD_09.01.02-53

ACTIONS

A.1

With a water level of < 23 ft above the top of the reactor vessel flange, all operations involving or movement of irradiated fuel assemblies within the containment shall be suspended immediately to ensure that a fuel handling accident cannot occur.

The suspension of fuel movement shall not preclude completion of movement of a component to a safe position.

A.2 DCD_09.01.02-53

With a water level of < 23 ft above the top of the reactor vessel flange, actions shall be initiated immediately to restore refueling cavity water level to within limits to provide adequate shielding and an adequate available heat sink.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.7.1

Verification of a minimum water level of 23 ft above the top of the reactor vessel flange ensures that the design basis for the analysis of the postulated fuel handling accident during refueling operations is met. Water at the required level above the top of the reactor vessel flange limits the consequences of damaged fuel rods that are postulated to result from a fuel handling accident inside containment (Ref. 2).

Verification of a minimum water level also ensures that sufficient water level exists to provide adequate shielding and an adequate available heat sink when one or more irradiated fuel assemblies are seated in containment racks. DCD_09.01.02-53

[The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls of valve positions, which make significant unplanned level changes unlikely. OR The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.]

REFERENCES

1. Regulatory Guide 1.183, July 2000.
2. Subsection 15.7.4.
3. 10 CFR 50.34

σ_i: 1σ for model considering tolerance i

2.3.1 Abnormal and Accident Conditions

The effects on reactivity of the credible abnormal and accident conditions are examined in this section. This section identifies which if any of the credible abnormal or accident conditions would result in exceeding the limiting reactivity (keff < 0.95). The double contingency principal (Reference [4]) specifies that it shall require at least two unlikely, independent and concurrent events to produce a criticality accident. This principle precludes the necessity of considering the simultaneous occurrence of multiple accident conditions.

2.3.1.1 Dropped Assembly – Horizontal

For the case in which a fuel assembly is assumed to be dropped on top of CR, the fuel assembly will come to rest horizontally on top of the rack with a minimum separation distance from the active fuel region of more than 17 inches, which is sufficient to preclude neutron coupling (i.e., an effectively infinite separation). Consequently, the horizontal fuel assembly drop accident will not result in an increase in reactivity and no separate calculation is performed for the drop accident.

2.3.1.2 Dropped Assembly – Vertical

It is also possible to vertically drop an assembly into a location that might be occupied by another assembly or that might be empty. The mechanical implications of such a drop been evaluated (Reference [9]). The result is shown in a loss of the top of the rack cells of 4 inches due to the vertical drop accident. However, CR are made of SS and do not contain fixed neutron absorber, thus the influence of the damage is of no concern to the criticality analyses.

2.3.1.3 Abnormal Location of a Fuel Assembly

2.3.1.3.1 Misloaded Fresh Fuel Assembly

Since CR are qualified for temporary storage of fresh fuel with the highest anticipated reactivity, the misloading of a fresh fuel assembly is of no concern.

2.3.1.3.2 Mislocated Fresh Fuel Assembly

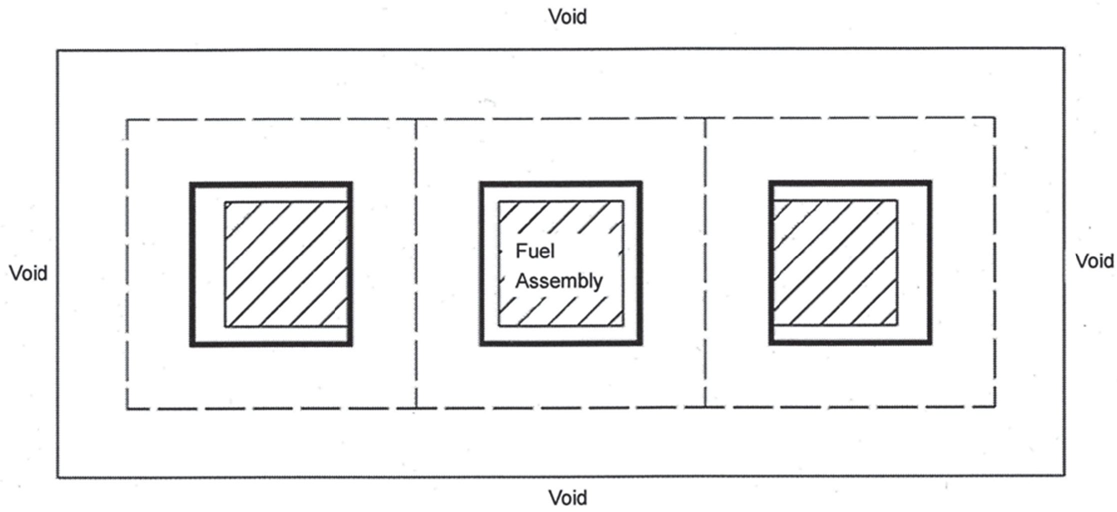
The mislocation of a fresh fuel assembly could, in the absence of soluble neutron absorber, result in exceeding the regulatory limit (keff < 0.95). This could possibly occur if a fresh fuel assembly of the maximum enrichment of five weight percent were to be accidentally mislocated outside of CR adjacent to other fuel assemblies. The CR could potentially have a fuel assembly mislocated outside its location and calculations were also performed for this case and the results were presented in Table 2-7. The amount of soluble boron needed to meet regulatory requirements was also determined by running the given accident case with 800 ppm soluble boron. This is far less than the boron concentration in the CR in the refueling cavity which is filled with borated water that has the refueling water storage pit boron concentration.

(Question 09.01.02-50, 51, 52)
Add:
and is controlled by Technical Specifications

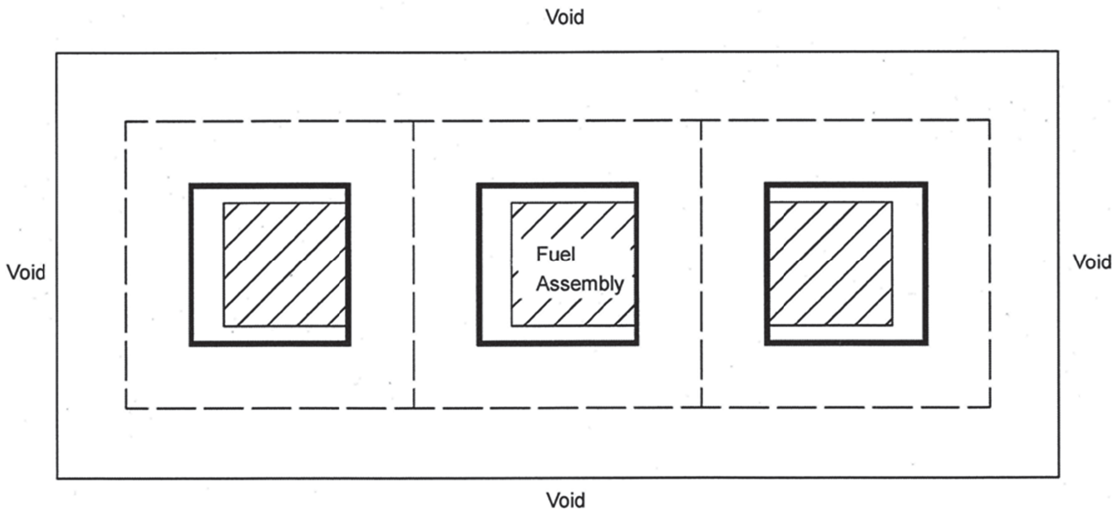
2.3.1.4 Rack Movement

The CR are fixed to the refu[illegible]e seismic force. Then the distance between two containment rack cells is not change after the seismic event.

(Question 09.01.02-59)
Add:
The analysis model is shown in Figure 2-5.



Eccentric Positioning Case 1



Eccentric Positioning Case 2

Figure 2-4 MCNP Model for Fuel Displacement within Cells of CR

(Question 09.01.02-59)
Add Figure 2-5 based on Figure 09.01.02-59.1.