KHNPDCDRAIsPEm Resource

From:	Ciocco, Jeff
Sent:	Thursday, August 20, 2015 10:43 AM
To:	apr1400rai@khnp.co.kr; KHNPDCDRAIsPEm Resource; Harry (Hyun Seung) Chang; Andy Jiyong Oh; Christopher Tyree
Cc:	Carlson, Donald; McKirgan, John; Wunder, George; Lee, Samuel
Subject:	APR1400 Design Certification Application RAI 167-8191 (09.01.01 - Criticality Safety of Fresh and Spent Fuel Storage and Handling)
Attachments:	APR1400 DC RAI 167 SRSB 8191.pdf; image001.jpg

KHNP,

The attachment contains the subject request for additional information (RAI). This RAI was sent to you in draft form. Your licensing review schedule assumes technically correct and complete responses within 30 days of receipt of RAIs. However, KHNP requests, and we grant, 60 days, 60 days, 60 days, 60 days, 30 days, 60 days, 60 days, 60 days, 60 days, 60 days, 90 days, and 60 days, for the 12 RAI questions. We may adjust the schedule accordingly.

Please submit your RAI response to the NRC Document Control Desk.

Thank you,

Jeff Ciocco
New Nuclear Reactor Licensing
301.415.6391
jeff.ciocco@nrc.gov

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

Hearing Identifier: KHNP_APR1400_DCD_RAI_Public
Email Number: 202

Mail Envelope Properties (bac04bd46ae0451cbaeb69be3400bc8a)

Subject: APR1400 Design Certification Application RAI 167-8191 (09.01.01 - Criticality Safety of Fresh and Spent Fuel Storage and Handling)
Sent Date: 8/20/2015 10:42:45 AM
Received Date: 8/20/2015 10:42:48 AM
From: Ciocco, Jeff

Created By: Jeff.Ciocco@nrc.gov

Recipients:
"Carlson, Donald" <Donald.Carlson@nrc.gov>
Tracking Status: None
"McKirgan, John" <John.McKirgan@nrc.gov>
Tracking Status: None
"Wunder, George" <George.Wunder@nrc.gov>
Tracking Status: None
"Lee, Samuel" <Samuel.Lee@nrc.gov>
Tracking Status: None
"apr1400rai@khnp.co.kr" <apr1400rai@khnp.co.kr>
Tracking Status: None
"KHNPDCDRAIsPEm Resource" <KHNPDCDRAIsPEm.Resource@nrc.gov>
Tracking Status: None
"Harry (Hyun Seung) Chang" <hyunseung.chang@gmail.com>
Tracking Status: None
"Andy Jiyong Oh" <jiyong.oh5@gmail.com>
Tracking Status: None
"Christopher Tyree" <Christopher.tyree@aecom.com>
Tracking Status: None

Post Office: HQPWMSMRS08.nrc.gov

Files	**Size**	**Date & Time**
MESSAGE	736	8/20/2015 10:42:48 AM
APR1400 DC RAI 167 SRSB 8191.pdf	141889	
image001.jpg	5040	

Options
Priority: Standard
Return Notification: No
Reply Requested: No
Sensitivity: Normal
Expiration Date:
Recipients Received:

REQUEST FOR ADDITIONAL INFORMATION 167-8191

Issue Date: 08/20/2015
Application Title: APR1400 Design Certification Review – 52-046
Operating Company: Korea Hydro & Nuclear Power Co. Ltd.
Docket No. 52-046
Review Section: 09.01.01 - Criticality Safety of Fresh and Spent Fuel Storage and Handling
Application Section:

QUESTIONS

09.01.01-2

RAI 9.1.1-4: Criticality prevention under normal and accident conditions of fuel handling

REQUIREMENTS AND GUIDANCE

In 10 CFR Part 50 Appendix A, General Design Criterion (GDC) 62 requires the prevention of criticality in fuel storage and handling. 10 CFR 50.68(b) sets specific requirements for the demonstration of nuclear criticality prevention in fuel storage. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 9.1.1, guides the reviewer to verify that the new and spent fuel will remain subcritical during fuel handling, in accordance with GDC 62 and 10 CFR 50.68(b).

Section 9.1.1 of NUREG-0800 also guides the reviewer to verify that the applicant has identified a comprehensive set of normal conditions and has modeled them conservatively. It also tells the reviewer that the accidental or erroneous placement of a fuel assembly outside of, but next to, the fuel storage racks should be considered as an abnormal condition. The NRC memorandum from Laurence Kopp to Timothy Collins, dated August 19, 1998, "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants," is also cited by the applicant and adds further clarification as follows: "However, by virtue of the double-contingency principle, two unlikely independent and concurrent incidents or postulated accidents are beyond the scope of the required analysis." With regard to the dry storage and handling of new fuel, the Kopp memorandum also notes that the accident conditions of flooding with optimum-density hydrogenous moderator (i.e., per 10 CFR 50.68(b)(3)) and flooding with full-density water (i.e., per 10 CFR 50.68(b)(2)) are the principal conditions that require evaluation and that the simultaneous occurrence of other accident conditions need not be considered.

As noted in the applicant-cited NRC Interim Staff Guidance DSS-ISG-2010-01, the normal conditions of fuel storage and handling include not only static storage but also anticipated fuel handling activities such as inspection, cleaning, reconfiguration, and movement of fuel in and around the fuel storage racks and associated fuel handling systems. Conditions during normal fuel handling operations are among the initial conditions to be considered in the analysis of postulated accidents.

ISSUE

It appears that the applicant has not provided information that demonstrates compliance with GDC 62 and 10 CFR 50.68(b) under normal and accident conditions of fuel handling in and around the various dry and wet fuel storage and handling systems. The wet fuel handling systems described in the DCD include those that handle fuel in and around the fuel elevator, the fuel transfer system, the refueling pool, the refueling cavity, and the spent fuel cask loading area. The staff notes for example that the fuel carrier of the fuel transfer system can hold two side-by-side fuel assemblies in an up-ended configuration and appears to have ample adjacent space into which an additional fuel assembly could be dropped by accident.

INFORMATION NEEDED

a) For dry handling of new fuel: In its response and in the DCD or its incorporated references, the applicant should identify a comprehensive set of normal conditions and accident conditions for dry handling of new fuel and provide conservative analyses that show compliance with GDC 62

and 10 CFR 50.68(b)(1)-(3) under such conditions. Where appropriate, the applicant should clairfy where administrative control procedures are relied upon to address this issue.

b) For wet handling of new and used fuel: In its response and in the DCD or its incorporated references, the applicant should identify a comprehensive set of normal conditions and accident conditions of wet fuel handling and provide information showing that all such handling conditions comply with GDC 62 and with 10 CFR 50.68(b)(4) where applicable. The information should conservatively evaluate the most reactive normal and accident conditions of fuel handling operations throughout the fuel storage and handling area of the auxiliary building and outside the reactor vessel in the reactor building. The wet fuel handling equipment and operations to be addressed include those that involve the following items as identified in the DCD: the pool racks, the fuel elevator, the fuel transfer system, the refueling canal, the refueling pool, the refueling cavity, the reactor cavity, and the cask loading area. Where appropriate, the applicant should clarify where administrative control procedures are relied upon to address this issue.

09.01.01-3

RAI 9.1.1-7: Detailed descriptions of fuel rack design configurations and interfaces

REQUIREMENTS AND GUIDANCE

In 10 CFR Part 50 Appendix A, General Design Criterion (GDC) 62 requires the prevention of criticality in fuel storage and handling. 10 CFR 50.68(b) sets specific requirements for the demonstration of nuclear criticality prevention in fuel storage. NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” Section 9.1.1, guides the reviewer, in part, to verify the completeness and appropriateness of fuel storage rack design data and their use in the analyses.

ISSUE

The application materials do not appear to include adequately detailed descriptions, such as plan section views and elevation of views, of the new and spent fuel storage rack designs. These design details are needed for the staff’s verification of DCD contents and for use in the staff’s independent confirmatory calculations. For example, the staff notes a lack of clear design information regarding the placement and positioning of absorber plates on storage cell walls, including in particular the four outer walls of the region I and region II spent fuel racks. Clear design information also appears to be lacking with regard to the relative alignment of outer-rack-wall-mounted absorber plates at the rack-to-rack interfaces within and between the two pool regions under normal and abnormal conditions.

INFORMATION NEEDED

In its response and in the DCD or its incorporated references, the applicant should provide adequately detailed descriptions, including plan section views and elevation of views, of the new and spent fuel storage racks and their respective configurations in the new fuel storage pit and spent fuel pool.

09.01.01-4

RAI 9.1.1-10: Modeled thickness of neutron absorber plates

REQUIREMENTS AND GUIDANCE

In 10 CFR Part 50 Appendix A, General Design Criterion (GDC) 62 requires the prevention of criticality in fuel storage and handling. 10 CFR 50.68(b)(4) sets specific requirements for the demonstration of nuclear criticality prevention in wet fuel storage. NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” Section 9.1.1, guides the reviewer, in part, to

verify that the criticality analysis conservatively incorporates fuel storage rack design data, including materials and dimensional data.

ISSUE

The applicant states that the absorber plates are assumed to have the maximum thickness allowed by tolerances. This staff considers this assumption to be potentially non-conservative in view of the lost neutron moderating effects of the water displaced by thicker plates.

INFORMATION NEEDED

In its response and in the DCD or its incorporated references, the applicant should resolve the apparent inconsistency by correctly characterizing and the modeling assumption of absorber plate thickness and by revising the modeling assumption as appropriate to avoid non-conservatism in this regard. If the assumption is revised, the applicant should also provide an updated criticality analysis for the storage pool racks.

09.01.01-5

RAI 9.1.1-11: Modeling of damaged fuel

REQUIREMENTS AND GUIDANCE

In 10 CFR Part 50 Appendix A, General Design Criterion (GDC) 62 requires the prevention of criticality in fuel storage and handling. 10 CFR 50.68(b)(4) sets specific requirements for the demonstration of nuclear criticality prevention in wet fuel storage. NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” Section 9.1.1, advises the reviewer, in part, to verify the conservatism of normal- and abnormal-conditions models and the appropriateness of assumptions and approximations made therein.

ISSUE

The design of the pool region I racks includes special rack cells for storing damaged fuel. The applicant has not characterized the allowed contents of those storage cells in terms of the assumed kinds and extent of fuel assembly damage or assembly reconfiguration analyzed for storage. As described in Section 3.4.2 and Figure 3.4-3 of the criticality analysis report, the applicant’s analysis model for the damaged fuel storage cells appears to contain an intact new fuel assembly. The NRC staff is concerned that this analysis model may not be bounding for the anticipated contents of damaged fuel and may be clearly bounding only for storing intact new fuel.

INFORMATION NEEDED

The applicant should provide in its response and in the DCD or its incorporated references a description of the allowed contents and conditions of fuel in the damaged fuel storage cells and, as necessary, an updated analysis with stated assumptions with regard to the modeling of allowed damaged fuel contents. If the analysis assumes immediate insertion of dummy or replacement fuel rods in place of rods damaged in the reactor or elsewhere, this should be stated and justified in the DCD or its incorporated references along with the identification of any supporting information or criteria to be provided in a COL.

09.01.01-6

RAI 9.1.1-12: Modeling of region I fuel rack outer walls

REQUIREMENTS AND GUIDANCE

In 10 CFR Part 50 Appendix A, General Design Criterion (GDC) 62 requires the prevention of criticality in fuel storage and handling. 10 CFR 50.68(b) sets specific requirements for the demonstration of nuclear criticality prevention in wet fuel storage. NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” Section 9.1.1, guides the reviewer, in part, to

verify that the materials of structures near racks that may provide neutron reflection, such as floors and walls, are provided and conservatively incorporated into the criticality analysis. Reviewers are further instructed to verify the conservatism of normal- and abnormal-conditions models and the appropriateness of assumptions and approximations made therein.

ISSUE

In Figure 3.4-3 of the criticality analysis report, it appears that not all cells along the outer walls of the racks in spent fuel pool region I have absorber plates facing the outer walls. It is therefore not clear that the infinite-array model shown in Figure 3.4-1 is conservative with regard to the finite-array effects of local moderation and reflection along the outer walls of the region I racks. In particular, the staff is concerned that the reactivity-increasing spectral effects of enhanced local neutron moderation and local thermalizing reflection in rack cells without absorber plates at their outer walls could in this case eventually prove to outweigh the reactivity-lowering leakage effects in the finite-array model such that the applicant's infinite-array reference model can no longer be seen as conservative.

INFORMATION NEEDED

In its response and in the DCD or its incorporated references, the applicant should provide a finite-array analysis that evaluates the infinite-array model's potential for non-conservatism with regard to modeling the spectral effects of enhanced nearby moderation and reflection in rack wall cells without absorber plates on the outer rack walls.

09.01.01-7

RAI 9.1.1-13: Reactivity effects of rodded fuel depletion

REQUIREMENTS AND GUIDANCE

In 10 CFR Part 50 Appendix A, General Design Criterion (GDC) 62 requires the prevention of criticality in fuel storage and handling. 10 CFR 50.68(b) sets specific requirements for the demonstration of nuclear criticality prevention in wet fuel storage. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 9.1.1, directs the reviewer to verify that appropriate assumptions are used in the criticality analysis. In addition, NRC Interim Staff Guidance DSS-ISG-2010-01 states that rodded operation may affect the discharge reactivity of fuel assemblies and should be considered in spent fuel pool criticality analyses. DSS-ISG-2010-01 further states that bounding reactor parameters should be used in the fuel depletion analysis.

ISSUE

The APR1400 nuclear design described in DCD Section 4.3 describes the use of inserted control rods at power and during load-follow operations, particularly part-strength rods. The staff notes that the effects of absorption and moderator displacement by inserted control rods substantially increase the reactivity of spent fuel by enhancing the production of fissile plutonium. Table 3.5-1 of the criticality analysis report lists the bounding reactor parameters for the depletion calculation, including parameters such as maximum fuel temperature and maximum fuel density, but the effects of rodded operation do not appear to be considered there or in other parts of the report. The neglect of rodded fuel depletion history is non-conservative.

INFORMATION NEEDED

In its response and in the DCD or its incorporated references, the applicant should either revise or justify the assumed fuel depletion parameters with regard to the effects if inserted control rods during depletion and provide revised depletion and criticality analyses as necessary.

09.01.01-8

RAI 9.1.1-14: Fuel power and temperature parameters in fuel depletion calculations

REQUIREMENTS AND GUIDANCE

In 10 CFR Part 50 Appendix A, General Design Criterion (GDC) 62 requires the prevention of criticality in fuel storage and handling. 10 CFR 50.68(b) sets specific requirements for the demonstration of nuclear criticality prevention in fuel storage. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 9.1.1, guides the reviewer to verify that the applicant has provided sufficient design and analysis information to support the evaluation findings. DSS-ISG-2010-01 states that bounding reactor parameters should be used in the fuel depletion analysis for spent fuel burnup credit. DSS-ISG-2010-01 further notes that it may be physically impossible for the fuel assembly to simultaneously experience two bounding values. In those cases, the applicant should maximize the dominant parameter and use the nominal value for the subordinate parameter. Where this is done, the application should describe and justify the parameters used. DSS-ISG-2010-01 also refers to the fuel depletion parametric studies described in NUREG/CR-6665, "Review and Prioritization of Technical Issues Related to Burnup Credit for LWR Fuel."

ISSUE

Table 3.5-1 of the criticality analysis report lists the bounding reactor parameters used in the fuel depletion calculations for fuel stored in the region II racks. The listed bounding reactor parameters include both a maximum fuel temperature and a maximum fuel power level, the latter being specified as a maximum fuel specific power in units of MWt/MTU. Although not clearly described as such in the report, the applicant's approach seems to attempt to credit the fact that these two parameters are correlated in such a way that it is not possible to operate a fuel assembly at conservative maximum fuel temperature and conservative minimum specific power at the same time.

The applicant's depletion sensitivity analysis for fuel specific power considers only a "nominal" power level versus an incrementally higher (i.e., ~13 percent higher) "maximum" power level. Similarly, the applicant's sensitivity analysis for fuel temperature considers only incremental variations of maximum fuel temperature. However, it is not clear to the staff that the applicant has considered the correlation between fuel power and fuel temperature in a consistent and appropriate manner. In order to evaluate whether fuel power or fuel temperature is the dominant parameter, it is important to compare the respective sensitivities on a consistent basis and over appropriate ranges of parameter values. Per DSS-ISG-2010-01, any conflicting correlation between reactivity maximizing and minimizing parameters should then be treated conservatively by setting the subordinate parameter to its nominal or expected value instead of to a minimizing value as otherwise might be done in a best-estimate approach to parameter correlation.

The staff notes that, per NUREG/CR-6665, longer operation at lower fuel power levels results in higher computed fuel reactivity, whereby fuel reactivity increases more when fuel power decreases late in life, which is typically the case. In terms of related considerations for axial burnup profiles, the staff notes that local fuel power levels are generally lower at the most reactive underburned ends of fuel assemblies. The staff also notes that DCD Section 4.3 refers to load-follow operations that would entail fuel depletion at reduced power levels. With regard to fuel temperature, NUREG/CR-6665 notes that higher fuel temperatures always increase spent fuel reactivity by enhancing fissile plutonium production.

The staff also notes that the applicant has not indicated how the listed values of expected and maximum fuel operating temperatures account for the thermal conductivity degradation discussed in NRC Information Notice 2009-03, "Nuclear Fuel Thermal Conductivity Degradation."

INFORMATION NEEDED

In its response and in the DCD or its incorporated references, the applicant should provide revised analyses that either (1) deplete the fuel at conservative values of high fuel temperature and low fuel power, thereby conservatively neglecting how the two parameters are correlated, or (2) deplete the fuel at a clearly conservative value of the dominant parameter and at a justified nominal or expected value of the subordinate parameter. If the second approach is taken, the supporting sensitivity analyses should be revised to provide a consistent basis for assessing parameter dominance and to address appropriately

lower ranges of fuel power (e.g., representing axial end effects and any allowed load-follow operations). The revised sensitivity results should then clearly show whether high fuel temperature or low fuel power is dominant for the expected ranges of operating parameters in this reactor design. For either approach, the revised analyses should also include an explanation of how the applicant's expected and maximum fuel temperature values account for fuel thermal conductivity degradation.

09.01.01-9

RAI 9.1.1-15: Determination of bounding axial burnup profiles

REQUIREMENTS AND GUIDANCE

In 10 CFR Part 50 Appendix A, General Design Criterion (GDC) 62 requires the prevention of criticality in fuel storage and handling. 10 CFR 50.68(b)(4) sets specific requirements for the demonstration of nuclear criticality prevention in wet fuel storage. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 9.1.1, directs the reviewer to verify that appropriate assumptions are used in the criticality analysis. In addition, NRC Interim Staff Guidance DSS-ISG-2010-01 notes that rodded operation can affect the axial burnup profiles and reactivity and states that conservative profiles should be identified and conservatively used in the analysis.

ISSUE

The criticality report states that bounding axial burnup profiles are selected by surveying a set of 304 burnup profiles that cover all possible types of axial burnup distributions. The staff needs to understand how that set of burnup profiles was generated (e.g., using on a comprehensive range of simulated reactor operating histories and/or by using core-follow calculations from similar operating plants) in order to verify that it indeed adequately covers the possible types of axial burnup distributions for this reactor design. To verify conservatism in the selection from that set of bounding burnup profiles from that set, the staff also needs to understand how the selection process considers the reactivity effects of local burnup history parameters such as rodded burnup.

INFORMATION NEEDED

In its response and in the DCD or its incorporated references, the applicant should provide information describing in detail how the set of axial burnup profiles was generated and how the set was evaluated and used to determine the most reactive bounding profiles. The requested information should also describe how the selection process considers the reactivity effects of local fuel depletion history parameters such as rodded burnup.

09.01.01-10

RAI 9.1.1-16: Uncertainty due to "fuel burnup measurement"

REQUIREMENTS AND GUIDANCE

In 10 CFR Part 50 Appendix A, General Design Criterion (GDC) 62 requires the prevention of criticality in fuel storage and handling. 10 CFR 50.68(b)(4) sets specific requirements for the demonstration of nuclear criticality prevention in wet fuel storage. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 9.1.1, guides the reviewer to verify that the applicant has provided sufficient design and analysis information to support the evaluation findings. As noted in NRC Interim Staff Guidance DSS-ISG-2010-01, the staff's reviews of the uses of burnup credit in spent fuel criticality safety analyses are informed by analytical studies presented in several NUREG/CR reports. NUREG/CR-6998, "Review of Information for Spent Fuel Burnup Confirmation," describes proposed measurement methods for confirming the recorded burnup of spent fuel assemblies as well as information on the accuracy of assembly burnup records absent such confirmatory measurements.

ISSUE

The applicant's criticality analysis report includes an analysis of uncertainty due to "fuel burnup measurement" but provides no indication that burnup measurements will be performed on the stored spent fuel assemblies. The applicant also cites information in NUREG/CR-6998 on the accuracy of spent fuel burnup records absent such confirmatory spent fuel measurements. To verify the appropriateness of the burnup uncertainty values used in the applicant's analysis, the staff needs to first understand whether the uncertainty analysis pertains to the burnup records or to spent fuel burnup measurements yet to be described. If the uncertainty pertains to burnup records, as is typically the case in today's operating plants, more information will be needed on the burnup record methods so that the staff can verify the appropriateness of the applicant's evaluation and treatment of fuel burnup uncertainty.

INFORMATION NEEDED

In its response and in the DCD or its incorporated references, the applicant should state whether burnup measurements will be performed on spent fuel assemblies and/or provide clarification as appropriate. If burnup measurements on spent fuel will in fact not be performed and the uncertainty in question pertains instead to burnup records, then the applicant should describe the burnup record methods with regard to how any nuclear design codes and/or in-core measurements with associated in-core analysis codes (i.e., codes identified in DCD Section 4.3, "Nuclear Design") are to be used in producing the fuel burnup records for APR1400 and what information the records contain (e.g., average burnup in the assembly as well as any information on assembly burnup profiles, burnup parameter histories such as rodded burnup, calculated nuclide compositions, etc.).

09.01.01-11

RAI 9.1.1-17: Adequacy of pool rack interface analyses

REQUIREMENTS AND GUIDANCE

In 10 CFR Part 50 Appendix A, General Design Criterion (GDC) 62 requires the prevention of criticality in fuel storage and handling. 10 CFR 50.68(b)(4) sets specific requirements for the demonstration of nuclear criticality prevention in wet fuel storage. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 9.1.1, guides the reviewer to verify that the applicant has provided sufficient design and analysis information to support the evaluation findings. NRC Interim Staff Guidance DSS-ISG-2010-01 states that the criticality analysis should consider the interfaces between storage configurations.

ISSUE

The applicant's criticality analysis report provides in Section 3.7 very brief descriptions of the rack interfaces within and between the pool regions. Lacking from the descriptions are adequately detailed illustrations of the racks and layouts that show interface details within and between the pool regions. The staff is therefore unable to verify the report's statements comparing the larger gap sizes between racks to the smaller gap sizes between rack cells in the reference models. Based on the limited dimensions and details that appear in the pool layout diagram in Figure 3.1-1 of the criticality analysis report, it appears that all gap sizes stated in Section 3.7 of the report are inflated by a factor of 10.

Apparent gap size discrepancies aside, the lack of detailed design and layout illustrations (e.g., plan section drawings at appropriate scales) makes it impossible for the staff to discern how the absorber panels on the outer rack walls align at the rack interfaces within pool region II. The staff is concerned that, should the interfacing wall absorber plates align such that some pairs of interfacing fuel assemblies have no absorber plates between them. This would then make it necessary for the applicant to explicitly model such interface configurations to determine if they can be more reactive than the applicant's region II reference model, which simply consists of an infinite-array rack model.

INFORMATION NEEDED

In its response and in DCD Section 9.1.1 or its incorporated references, the applicant should correct all factual errors (such as misstated gap dimensions) and provide detailed design information, such as appropriately scaled plan section drawings, that shows the gaps between racks within and between the pool regions. The interface design drawings should show in detail how the wall absorber plates align at the rack interfaces within region II. If unmeshed absorber plate alignments at rack interfaces are expected or possible under normal or accident conditions, the information should include explicitly modeled analyses of those interface configurations.

09.01.01-12

RAI 9.1.1-18: Additional trending parameters for criticality code bias and uncertainty

REQUIREMENTS AND GUIDANCE

In 10 CFR Part 50 Appendix A, General Design Criterion (GDC) 62 requires the prevention of criticality in fuel storage and handling. 10 CFR 50.68(b) sets specific requirements for the demonstration of nuclear criticality prevention in fuel storage. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 9.1.1, guides the staff to review the computational method validation to verify that the validation study is thorough and uses benchmark critical experiments that are similar to the normal-conditions and abnormal conditions models and to verify that the neutron multiplication factor's bias and bias uncertainty values are conservatively determined.

ISSUE

The applicant's criticality validation report describes results from benchmarking the applicant's criticality code against sets of data from laboratory critical experiments performed on applicably similar water moderated arrangements of absorbers and arrayed fresh fuel rods of various uranium enrichments or mixed oxide compositions. The resulting code k-eff biases were statistically analyzed by the applicant against several postulated trending parameters with the conclusion that only uranium enrichment exhibits statistically significant bias and uncertainty trends. However, the trending parameters considered by the applicant did not include plutonium content or plutonium fission fraction. The NRC staff is aware of similar code validation benchmark studies that have found statistically significant trends against plutonium content. The staff is therefore concerned that applicant's determination of code bias and bias uncertainty has not conservatively addressed the code bias and uncertainty associated with plutonium effects.

INFORMATION NEEDED

In its response and in the DCD or its incorporated references, the applicant should provide a supplemental trending study of its experimental code benchmark results against plutonium content and/or code-computed plutonium fission fraction. If the applicant finds that the selected benchmarks with plutonium are too few for these purposes, then the applicant should state so and pursue additional benchmarks for addressing this issue. If supplemental trending results show statistically significant code bias trends against plutonium parameters such that more conservative bias and uncertainty adjustments would result, then the applicant should revise the code validation report accordingly and apply the resulting more conservative bias and uncertainty adjustments to its computed criticality results.

09.01.01-13

RAI 9.1.1-25: Abnormal conditions identification and development

REQUIREMENTS AND GUIDANCE

In 10 CFR Part 50 Appendix A, General Design Criterion (GDC) 62 requires the prevention of criticality in fuel storage and handling. 10 CFR 50.68(b) sets specific requirements for the demonstration of nuclear

criticality prevention in fuel storage and handling. NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition” (SRP), Section 9.1.1 guides the reviewer, in part, to verify that the scope of considered abnormal conditions is comprehensive. A description of the process used to identify abnormal conditions should be provided and should include consideration of common-mode failures. In addition, the SRP Acceptance Criteria for Section 9.1.1 refer to ANSI/ANS-57.2 and ANSI/ANS-57.3. ANSI/ANS-57.2 states that the criticality safety analysis for spent fuel storage racks should include consideration of credible abnormal occurrences; a few examples include tipping or falling of a spent fuel assembly, tipping of a storage rack, fuel drop accidents, and horizontal movement of fuel before complete removal from the rack. ANSI/ANS-57.3 similarly lists credible accidents and conditions for new fuel racks.

ISSUE

The criticality technical report describes and analyzes postulated accidents for the new and spent fuel storage racks, including moderation events for the new fuel racks and, for the spent fuel racks, a vertically dropped fresh fuel assembly outside of the racks, a misloaded fresh fuel assembly, and a boron dilution accident. However, the technical report does not describe how the abnormal conditions were identified or how other potential accidents listed in ANSI/ANS-57.2 and ANSI/ANS-57.3 are bounded by existing analyses.

INFORMATION NEEDED

In its response, the applicant should describe the logic employed when identifying the abnormal conditions to analyze in its criticality analysis, including how credible events such as tipping fuel assemblies or racks and dropped fuel assemblies are bounded by existing analyses.

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment