AP1000DCDFileNPEm Resource

From: Adams II, Samuel L. [adamssl@westinghouse.com]
Sent: Thursday, May 08, 2008 1:50 PM
To: Phyllis Clark
Cc: Perry Buckberg; Rhonda Carmon
Subject: FW: RAI Transmittal
Attachments: Ch. 4 & 15 RAIs - SRSB.doc

Hi Phyllis,

I acknowledge receipt of the attached RAIs on Ch4 and Ch15.

I will let you know as soon as possible if a clarification call is necessary.

Thanks.

Sam

From: Phyllis Clark [mailto:Phyllis.Clark@nrc.gov]
Sent: Thursday, May 08, 2008 9:34 AM
To: Adams II, Samuel L.; Lindgren, Donald A.
Subject: RAI Transmittal

Hi Sam,

Attached are 17 RAIs associated with the review of AP1000 DCD Revision 16, Chapters 4 and 15. The RAI's are numbered as follows in the attachment:

RAI-SRP4.1-SRSB-01
RAI-SRP4.2-SRSB-01
RAI-SRP4.2-SRSB-02
RAI-SRP4.2-SRSB-03
RAI-SRP4.2-SRSB-04
RAI-SRP4.2-SRSB-05
RAI-SRP4.2-SRSB-06
RAI-SRP4.3-SRSB-01
RAI-SRP4.3-SRSB-02
RAI-SRP4.4-SRSB-01
RAI-SRP4.4-SRSB-02
RAI-SRP15.0-SRSB-01
RAI-SRP15.1.4-SRSB-01
RAI-SRP15.3.1-SRSB-01
RAI-SRP15.4.3-SRSB-01
RAI-SRP15.4.6-SRSB-01
RAI-SRP15.5.1-SRSB-01

Please provide me an e-mail confirming your receipt of the attached RAIs.

P. Clark
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Washington, DC 20555
301-415-6447
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Hearing Identifier: AP1000_DCD_Review
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Subject: FW: RAI Transmittal
Sent Date: 5/8/2008 1:50:20 PM
Received Date: 5/8/2008 1:50:23 PM
From: Adams II, Samuel L.

Created By: adamssl@westinghouse.com

Recipients:
"Perry Buckberg" <Perry.Buckberg@nrc.gov>
Tracking Status: None
"Rhonda Carmon" <Rhonda.Carmon@nrc.gov>
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Tracking Status: None

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MESSAGE	1216	5/8/2008 1:50:23 PM
Ch. 4 & 15 RAIs - SRSB.doc	37440	

Options
Priority: Standard
Return Notification: No
Reply Requested: No
Sensitivity: Normal
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REQUEST FOR ADDITIONAL INFORMATION (RAI) FOR AP1000 DCD REVISION 16
(Chapter 4 and 15)

RAI - SRP 4.1 - SRSB - 01

Table 4.1-1, "Reactor Design Comparison Table," indicates a design change to include neutron panels in the reactor internals design. The addition of four neutron panels attached to the outside surface of the core barrel, along with the addition of a flow skirt in the lower reactor vessel head, is described in TR 29, WCAP-16716-NP, Revision 2, "AP1000 Reactor Internals Design Changes." The NRC staff previously requested that Westinghouse provide an evaluation of the impacts of these proposed changes on the analysis results of each of the transients and accidents in DCD Chapter 15. The evaluation should demonstrate either the analysis results of DCD Chapter 15 remain bounding and valid, or the effects of these changes are insignificantly small that the applicable acceptance criteria for these transients and accidents remain complied with. In its letter of January 3, 2008, Westinghouse responded to the NRC request for additional information (RAI), indicating that it will provide an evaluation to demonstrate that the reactor internals change does not significantly impact the DCD Chapter 15 analysis results, and that the evaluation will be provided as a revision to the RAI response in April 2008 time frame.

Please provide the evaluation results as indicated in your January 3, 2008, letter, if not already provided.

RAI - SRP 4.2 - SRSB - 01

Revision 16 of DCD Chapter 4 reclassifies the AP1000 core and fuel design Tier 2* items to Tier 2. TR 119 (APP-GW-GLR-119, Revision 0) is used to support the reclassification of the DCD Chapter 4 Tier 2* items to Tier 2. The basis for this change is the commitment to supply a future topical report, called the core reference report. While the submittal of a core reference report satisfies the Tier 2* requirement of NRC review of any changes to the DCD Tier 2* items, there is no support as to why it is appropriate to also reclassify these Tier 2* items themselves. It is contradictory to the original intent of making these items Tier 2* in that by reclassifying them to Tier 2, it appears changes could later be made to the submitted core reference report or even the DCD material without prior NRC approval.

Explain how the proposed submittal (core reference report) ensures that future changes for those items originally classified as Tier 2* will be reviewed and approved by the NRC?

RAI - SRP 4.2 - SRSB - 02

Section 4.2.1.6.2 changes the burnable absorber rods from a Wet Annular Burnable Absorber (WABA) to a borosilicate glass design. WCAP-7113 is referenced as justification of this change, but this document does not demonstrate prior NRC approval.

(a) Please provide proof of NRC approval of the borosilicate glass design.
(b) Please provide supporting documentation that demonstrates the applicability of borosilicate glass burnable absorbers to planned AP1000 conditions, including any operational experience and any post-irradiation examination (PIE) data.

RAI - SRP 4.2 - SRSB - 03

Section 4.2.1.6.2 lists multiple burnable absorber designs. Please only include the planned design for the initial core.

RAI - SRP 4.2 - SRSB - 04

Section 4.2.2.1 deletes an erroneous reference for the Integral Fuel Burnable Absorber (IFBA) design. Please provide the correct reference instead of only removing the old reference.

RAI - SRP 4.2 - SRSB - 05

Revision 16 of subsection 4.2.2.3.2 changes the design of the Gray Rod Control Assemblies (GRCAs). While it is recognized that the GRCAs are used for load follow maneuvering and are not credited as a safety system, the changes in the GRCA design may have impacts on safety analyses. In particular, can the GRCA design changes result in a more limiting case than the accident analyses described in Chapter 15, such as the rod ejection accident analysis?

RAI - SRP 4.2 - SRSB - 06

Revision 16 of subsection 4.2.5 references APP-GW-GLR-059 (Reference 25). Reference 25 is listed as APP-GW-GLR-059, "Changes to Reference Reactor Design," and Reference 24 is listed as WCAP-16652-NP, "AP1000 Core & Fuel Design Technical Report," Revision 0. However, both APP-GW-GLR-059 and WCAP-16652-NP, Revision 0, appear on the cover of the AP1000 Core & Fuel Design Technical Report, indicating they are the same report.

Clarify the inconsistencies in subsection 4.2.5 and associated references 24 and 25. The same inconsistencies exist in subsection 4.3.4 and associated references 64 and 65.

RAI - SRP 4.3 - SRSB - 01

In Section 4.3.2.2.6, "Limiting Power Distributions," it is stated that, "the online monitoring system is not a required element for short term reactor operation".

Please discuss the proposed time frame ("short term") that the online monitoring outage is not a requirement for operation.

RAI - SRP 4.3 - SRSB - 02

DCD Section 4.3.2.4.2 deleted specific temperature increase value ($4^{o}F$) from the sentence that the design change in temperature is conservatively increased by $4^{o}F$ to account for the control system dead band and measurement errors. TR 18 (WCAP-16650-NP, Revision 0, APP-GW-GLR-059) provided the rationale for this change.

Please explain how the dead band and measurement errors are now calculated.

RAI - SRP 4.4 - SRSB - 01

Revision 16 of DCD Section 4.4, Reference 87 lists WCAP-16652-NP, "AP1000 Core & Fuel Design Technical Report," Revision 0, which is not referenced in the DCD Section 4.4 text. Revision 16 of DCD Subsection 4.4.7, Combined License Information remains unchanged from Revision 15, contradictory to what was shown in WCAP-16652-NP.

Please clarify the inconsistency.

RAI – SRP 4.4 –SRSB – 02

DCD Subsection 4.4.7 states that "Following selection of the actual plant operating instrumentation and calculation of the instrumentation uncertainties of the operating plant parameters as discussed in subsection 7.1.6, Combined License applicants will calculate the design limit DNBR using the RTDP with these instrumentation uncertainties …." In Revision 16 of DCD, subsection 7.1.6 has been revised to state that "The Combined License information requested in this subsection has been completely addressed in WCAP-16361-P (Reference 17), and …. No additional work is required by the Combined License applicant."

Please clarify the apparent inconsistency in DCD subsections 4.4.7 and 7.1.6 regarding the need for COL applicants to calculate the actual instrumentation uncertainties of the plant operating parameters. Identify the methodology that will be used for the calculation of instrumentation uncertainties of the operating plant parameters as part of design limit DNBR calculation, and explain why this methodology is acceptable.

RAI - SRP 15.0 - SRSB - 01

In Revision 16 of AP1000 DCD Table 15.0-4a, the limiting setpoint assumed in the safety analysis for the source range neutron flux doubling (or multiplication) for boron dilution block is changed from 1.6 to 2.2 over 50 minutes. The 2.2/50 minutes setpoint is the same as the nominal setpoint value in revision 16 of TS Table 3.3.2-1, ESFAS function 15, Boron Dilution Block. In TR 80 (APP-GW-GLR-080, Revision 0), it is stated that a reanalysis of the postulated decrease in the RCS boron concentration during shutdown event (DCD section 15.4.6) has been performed and it has verified that an increase in the nominal setpoint from 1.6 to 2.2 demonstrates acceptable results.

a. Provide the value of the source range flux doubling setpoint assumed in the safety analysis of Section 15.4.6, Revision 16.

b. Explain how the safety analysis setpoint value is consistent with the TS setpoint value of 2.2/50. The explanation should include a detailed description of the instrumentation uncertainties associated with the calculation of the source range neutron leakage count rate, and how these uncertainties are accounted for in the determination of the TS setpoint and allowable value.

RAI - SRP 15.1.4 - SRSB - 01

In DCD subsections 15.1.4 and 15.1.5, the following changes are made (1) the core makeup tank actuation signals is changed from "two of out four low pressurizer level signals" to "two of out four low-2 pressurizer level signals; and (2) the trip signal of the fast-acting main steam line isolation valves is changed from "two out of four high containment pressure signals" to "two out of four high-2 containment pressure signals."

Please confirm that (1) these changes are made merely to be consistent with the corresponding trip functions in TS Table 3.3.2-1, and (2) the actual setpoints used in the safety analyses of these events remain the same and are not affected by the changes.

RAI - SRP 15.3.1 - SRSB - 01

Revision 16 of DCD Table 15.0-6 shows that the power range neutron flux P-8 permissive interlock for low flow trip function is deleted and replaced by P-10 interlock. As explained in TR 80, the deletion of P-8 interlock is based on the fact that AP1000 is not approved for N-1 loop operation.

Since the trip setpoint values for P-8 [48%] and P-10 [10%] are quite different, please confirm that the safety analyses of transient events in DCD Revision 16 (such as loss of flow in DCD subsections 15.3.1 and 15.3.2) that take credit of the P-8 or P-10 interlock have been re-analyzed to reflect this change (from P-8 to P-10 interlock)?

RAI - SRP 15.4.3 - SRSB -01

For DCD revision 16, a change is made to subsection 15.4.3.1 by deleting the statement that if the rod deviation alarm is not operable, the operator takes action as required by the Technical Specifications.

If operator action is required by TS in the event that one or more of the rod position indicator channels is out of service, what is the basis for deleting the operator action in the sentence in DCD subsection 15.4.3.1? What ensures that the rod deviation alarm remains operable at all times so that operator action should no longer be required by TS?

RAI - SRP 15.4.6 - SRSB - 01

On DCD pages 15.4-20 and 15.4-21 (subsection 15.4.6) a change is made to delete the source range nuclear instrumentation detection value of “an increase of 60 percent” with generic text reading “a significantly large increase.” Page 59 of APP-GW-GLR-080, Revision 0 discusses the increase of 60 percent but does not mention replacing the 60 percent value in the text.

Given that an increase of 60 percent is a measureable and detectable value, please define what a “sufficiently large” increase means and how it is calculated.

RAI - SRP 15.5.1 - SRSB - 01

In the existing DCD subsection 15.5.1.3, it was stated that the limiting case presented here bounds all cases that model explicit operator action 30 minutes after a reactor trip. Revision 16 changes the operator action from 30 minutes to 60 minutes after a reactor trip. However, the figures and tables provided in Subsection 15.5.1 do not appear to be updated or revised.

Is operator action required to mitigate this transient? If so, what is the effect of delaying the operator action time by 30 minutes on the transient response? Have the events been re-analyzed? Have the results in revision 16 of DCD 15.5.1 reflected the revised analysis

results, figures, and tables related to the effects of delaying the operator action time from 30 minutes to 60 minutes?