ArevaEPRDCPEm Resource

From: Getachew Tesfaye
Sent: Tuesday, March 24, 2009 1:10 PM
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Subject: U.S. EPR Design Certification Application RAI No. 195 (2259)), FSAR Ch. 4
Attachments: RAI_195_SRSB_2259.doc

Attached please find the subject requests for additional information (RAI). A draft of the RAI was provided to you on February 26, 2009, and discussed with your staff on March 11, 2009. RAI Question 04.02-4 was modified as a result of that discussion. The schedule we have established for review of your application assumes technically correct and complete responses within 30 days of receipt of RAIs. For any RAIs that cannot be answered within 30 days, it is expected that a date for receipt of this information will be provided to the staff within the 30 day period so that the staff can assess how this information will impact the published schedule.

Thanks,
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Hearing Identifier: AREVA_EPR_DC_RAIs
Email Number: 331

Mail Envelope Properties (C56E360E9D804F4B95BC673F886381E71DBDFDC7CC)

Subject: U.S. EPR Design Certification Application RAI No. 195 (2259)), FSAR Ch. 4
Sent Date: 3/24/2009 1:09:38 PM
Received Date: 3/24/2009 1:09:39 PM
From: Getachew Tesfaye

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Post Office: HQCLSTR02.nrc.gov

Files	**Size**	**Date & Time**
MESSAGE	759	3/24/2009 1:09:39 PM
RAI_195_SRSB_2259.doc	44538	

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

Request for Additional Information No. 195 (2259), Revision 0

03/24/2009

U. S. EPR Standard Design Certification
AREVA NP Inc.
Docket No. 52-020
SRP Section: 04.02 - Fuel System Design
Application Section: 4.2

QUESTIONS for Reactor System, Nuclear Performance and Code Review (SRSB)

04.02-3

FSAR Section 4.2.1.2.2 states that fuel densification and fission product swelling are addressed in AREVA NP topical report BAW-10231P-A, "COPERNIC Fuel Design Computer Code," January 2004. The topical report provides a description of the fuel densification and swelling model along with COPERNIC analysis results and comparison to measured data. Fuel densification (with solid swelling) results are presented for the Integrated Dry Route (IDR), Ammonium Di-Uranate (ADU) wet, and Ammonium Uranyl Carbonate (AUC) wet conversion processes.

In order for the staff to complete its evaluation of the performance of the U.S. EPR fuel against the requirements of GDC 10 as it relates to SAFDLs for normal operation and during AOOs, analysis results specifically applicable to the U.S. EPR fuel are required.

Provide evaluation results showing the fuel densification (with solid swelling) as a function of fuel rod burnup for the U.S. EPR fuel, including the appropriate initial pellet density (% TD) and uranium material conversion process. Assess the effect of fuel pellet Gd_2O_3 content up to the 8 weight percent design limit.

04.02-4

Section 4.2.1.3.2 of the FSAR states that the possibility of fuel rod cladding creep collapse is precluded by high theoretical density as-fabricated pellets which exhibit very low densification during operation and by manufacturing process inspections to prevent fuel column gaps.

ANP-10285P, "U.S. EPR Fuel Assembly Mechanical Design Report" addresses fuel rod cladding collapse, referencing an NRC-approved methodology (BAW-10227P-A, "Evaluation of Advanced Cladding and Structural Material (M5TM) in PWR Reactor Fuel") and stating that the creep rate for the M5TM cladding is approximately 50% slower than for Zircaloy-4 and that the fuel rod creep collapse lifetime is greater than the fuel rod design burnup limit of 62 GWD/MTU. NRC staff's February 4, 2000 safety evaluation of BAW-10227P-A, "Evaluation of Advanced Cladding and Structural Material (M5TM) in PWR Reactor Fuel," however, discusses a disparity between the earlier Zircaloy-4 creep model (with M5TM adjustment factor) and a new M5TM creep model.

In order for the staff to complete its evaluation of the performance of the U.S. EPR fuel against the requirements of GDC 10 as it relates to SAFDLs for normal operation and during AOOs, additional information is required.

Provide the following information relative to the above discussion on fuel rod cladding creep:

a. Clarify which creep model is applied to the U.S. EPR fuel, and verify model applicability to the fuel rod burnup range from 0 to 62 GWD/MTU.
b. Confirm, clarify, or revise the statement given in ANP-10285P, "U.S. EPR Fuel Assembly Mechanical Design Report" that the M5™ creep rate is 50% slower than for Zircaloy-4.
c. Confirm the conclusion that the U.S. PER fuel rod creep collapse lifetime is greater than the fuel rod design burnup limit of 62 GWD/MTU.

04.02-5

In order for the staff to complete its evaluation of the U.S. EPR fuel assembly structural design relative to the requirements of GDC 10, additional information regarding nonoperational loading of the U.S. EPR fuel assembly is required.

Identify the nonoperational loads, and provide the basis for the axial and lateral loads that are given in FSAR Section 4.2.1.5.1.

04.02-6

In order for the staff to complete its evaluation of the U.S. EPR fuel assembly structural design relative to the requirements of GDC 27 as it relates to the capability of the reactivity control systems, additional information is required.

Justify the use of maximum shear stress theory to evaluate the U.S. EPR fuel assembly guide tube design and provide the evaluation results for the U.S. EPR fuel assembly guide tube.

04.02-7

Section 4.2.1.5.5 of the FSAR states that the design bases for fuel assembly holddown springs is to maintain fuel assembly contact with the lower core plate, except during a reactor coolant pump overspeed transient. This does not meet the design acceptance criterion of SRP Section II.1.A.vii. The staff notes, however, that its safety evaluation of BAW-10239P-A, Revision 0, "Advanced Mark-BW Fuel Assembly Mechanical Design Topical Report" found the pump overspeed transient exception acceptable for the advanced Mark-BW design.

Additional information is required in order for the staff to complete its evaluation of the U.S. EPR control rod (RCCA) design relative to the requirements of GDC 27 and the guidelines of SRP Section II.1.A.viii as they relate to insertability of control rods.

Justify applicability of the above-described exception to the SRP Section II.1.A.vii acceptance criterion for the U.S. EPR fuel design with the bottom FUELGUARD™ nozzle. Specifically, assess the effect of the FUELGUARD™ nozzle on assembly liftoff.

04.02-8

Sections 4.2.1.7 and 4.2.4.6 describe the post-irradiation surveillance program for the U.S. EPR fuel and the RCCAs.

In order for the staff to complete its evaluation of the performance of the U.S. EPR fuel against the requirements of GDC 10 and relative to the guidelines of SRP Section II Acceptance Criteria 4.C. as they relate to post-irradiation surveillance of the U.S. EPR fuel, additional information is required.

What actions are taken when: (1) plant instrumentation detects a fuel rod failure, and (2) where post-irradiation fuel examinations indicate unusual fuel system behavior. Include a description of the actions that may be taken, the disposition of failed fuel, and the root cause analyses that may be performed.

04.02-9

FSAR Table 4.2-1 provides a summary of the U.S. EPR fuel assembly design and lists the fissile enrichment to be less than or equal to 4.95 weight percent U^{235}. Section 3.2 of ANP-10285P, "U.S. EPR Fuel Assembly Mechanical Design Topical Report" states that pellet enrichments used for the U.S. EPR may be as high as 5 weight percent U^{235}.

FSAR Figure 4.2-2 shows the U.S. EPR fuel rod assembly with a support tube to end of fuel rod dimension of 5.315 inches. Figure 3-3 of ANP-10285P, "U.S. EPR Fuel Assembly Mechanical Design Topical Report" shows the same parameter dimension to be 2.048 inches.

Resolve the noted discrepancies and correct the documents as appropriate.

04.02-10

Section 4.2.2.9 describes a control rod end plug flex joint, designed to accommodate misalignment of the RCCA control rods with the fuel assembly guide tubes.

In order for the staff to complete its evaluation of the U.S. EPR control rod (RCCA) design relative to the requirements of GDC 27 and the guidelines of SRP Section II.1.A.viii as they relate to insertability of control rods, additional information is required.

Provide the following additional information on the control rod flex joint:

a. Description and drawings, explaining the flex joint connection to the RCCA spider, the flex joint material, and how it functions to accommodate misalignment of the RCCA control rod with the fuel assembly guide tubes.
b. Discuss the operational experience with the control rod flex joint design feature, including any failures of the flex joint.

04.02-11

FSAR Table 4.2-4 and Figure 4.2-12 provide Rod Control Cluster Assembly (RCCA) design data. The dimensions given for the hub diameter and the total height differ slightly between the table and the figure.

Correct the discrepancy and confirm that all the data provided in Table 4.2-4 and Figure 4.2-12 are correct.

04.02-12

Section 4.2.3.1.4 states that crevice corrosion is not a likely corrosion mechanism for zirconium alloy material and that in general, zirconium alloys are very resistant to crevice corrosion.

In order for the staff to complete its evaluation of the performance of the U.S. EPR fuel against the requirements of GDC 10 as it relates to SAFDLs for normal operation and during AOOs, the following information is requested.

Justify the statements concerning M5™ fuel rod cladding resistance to crevice corrosion and provide the following information:

a. A description of post irradiation examination results of applicable PWR fuel relative to crevice corrosion.
b. An explanation of applicability of the fuel examination results to the M5™ fuel rod cladding used in the U.S. EPR fuel system design.

04.02-13

Section 4.2.3.2.4 states that irradiation stability of the U.S. EPR fuel pellet is confirmed by performing analyses using the COPERNIC code.

In order for the staff to complete its evaluation of the performance of the U.S. EPR fuel against the requirements of GDC 10 as it relates to SAFDLs for normal operation and during AOOs, additional information is required.

Provide a description and the results of the COPERNIC analyses that were performed to confirm the irradiation stability of the U.S. EPR fuel pellet.

04.02-14

Section 4.2.3.6.1 states that calculations of RCCA rod internal gas pressure and cladding stresses have been performed and that for all the cases analyzed the cladding stresses remain within acceptable limits. Similarly, the neutron source rod cladding stresses are stated to remain below the stress limits.

Additional information is required in order for the staff to complete its evaluation of the U.S. EPR fuel control rod (RCCA) design relative to the requirements of GDC 27 as it relates to the capability of the reactivity control systems. The performance of the neutron source tubes must be evaluated relative to their potential effect on surrounding fuel rods relative to GDC 10.

Explain the source of control rod pressurization during normal operation, AOOs, and accident (LOCA) conditions.

What are the stress level acceptance criteria for conditions of normal operation, AOOs, and accident (LOCA) conditions, and how do the calculated internal control rod pressures and stress levels compare to the criteria?

Provide the analysis results for the calculated internal rod pressures and stresses with comparison to the acceptance criteria.

04.02-15

Section 4.2.4.1.4 states that more than 4,600 ion-nitrated HARMONI™ RCCAs have been delivered to operating reactors, including 572 RCCAs to U.S. PWRs since 1995.

Additional information is required in order for the staff to complete its evaluation of the U.S. EPR control rod design relative to the requirements of GDC 27 and the guidelines of SRP Section II.1.A.viii as they relate to the capability of the reactivity control systems.

Provide additional information on the design and performance experience of the HARMONI™ RCCAs relative to the U.S. EPR RCCA designs, including:

a. Identification of reactor facility, number of RCCAs delivered, and year of installation.
b. Any difference in the mechanical design, configuration, or materials of the delivered RCCAs relative to the U.S. EPR RCCAs.
c. Identification of any failures or degradations of the RCCAs.