ArevaEPRDCPEm Resource

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Sent:	Monday, December 15, 2008 7:51 AM
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Cc:	John Budzynski; Shanlai Lu; Joseph Donoghue; Jason Carneal; Prosanta Chowdhury; Joseph Colaccino; John Rycyna; ArevaEPRDCPEm Resource
Subject:	U.S. EPR Design Certification Application RAI No. 134 (1435, 1279, 1436), FSAR Ch. 4
Attachments:	RAI_134_SRSB_1435_1279_1436.doc

Attached please find the subject requests for additional information (RAI). A draft of the RAI was provided to you on November 6, 2008, and discussed with your staff on November 19, 2008. Draft RAI Questions 04.04-28 and 04.06-7 were deleted, Draft Questions 04.04-32, 04.04-35, 04.06-4, 04.06-5, and 04.06-8 were moved to other FSAR chapters, and Draft Questions 04.03-10, 04.06-3, 04.06-9, and 04.06-10 were 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, excluding the time period of **December 20, 2008 thru January 1, 2009, to account for the holiday season** as discussed with AREVA NP Inc. For any RAIs that cannot be answered **within 45 days**, it is expected that a date for receipt of this information will be provided to the staff within the 45-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: 41

Mail Envelope Properties (C56E360E9D804F4B95BC673F886381E71827CBAEC7)

Subject: U.S. EPR Design Certification Application RAI No. 134 (1435, 1279, 1436), FSAR Ch. 4
Sent Date: 12/15/2008 7:50:32 AM
Received Date: 12/15/2008 7:50:43 AM
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Files	**Size**	**Date & Time**
MESSAGE	1103	12/15/2008 7:50:43 AM
RAI_134_SRSB_1435_1279_1436.doc	80378	

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

Request for Additional Information No. 134 (1435, 1279, 1436), Revision 0

12/15/2008

U. S. EPR Standard Design Certification
AREVA NP Inc.
Docket No. 52-020
SRP Section: 04.03 - Nuclear Design
SRP Section: 04.04 - Thermal and Hydraulic Design
SRP Section: 04.06 - Functional Design of Control Rod Drive System
Application Section: FSAR Ch. 4

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

04.03-7

FSAR Tier 2 Section 4.3.1.1 describes the fuel burnup limit in terms of a peak fuel rod burnup limit of 62 GWD/MTU. However, FSAR Table 4.3-1 lists the burnup limit for gadolinia fuel rods as 55 GWD/MTU.

Provide additional information on the 55 GWD/MTU design peak rod burnup for gadolinia fuel rods, including technical rationale and basis, its application, and its effect on the assembly burnup limit.

This information is needed in order to clarify and ensure the adequacy of the U.S. EPR fuel burnup limit, and further to ensure that the design will be consistent with GDC10.

04.03-8

FSAR Tier 2 Figure 4.3-6 through Figure 4.3-9 show the placement of fuel rods within the seven different fuel assembly designs of the initial core. Fuel assembly designs B1, B2, C2, and C3 contain gadolinia fuel rods that are located face-adjacent to water-filled guide tubes. Such placement of gadolinia-bearing fuel rods could result in preferential uneven depletion of the gadolinia, thus posing a challenge to the lattice physics model calculations and hence introduction of modeling error.

Provide justification for the placement of gadolinia-bearing fuel rods face-adjacent to the water-filled guide tubes, including description of the validation of the MICBURN-3/CASMO-3G codes for such design. Benchmark results or reference to applicable licensing topical reports should be provided.

This information is required in order to ensure that the analytical methods utilized in the design of the U.S. EPR fuel assembly are capable of accurately modeling the design and that they are thus consistent with GDC 10.

04.03-9

FSAR Tier 2 Section 4.3.2.1 states that the core average enrichment is set by the fuel cycle length and energy requirements.

Explain the rationale for that statement and state if and how the fuel assembly design burnup limit, which is based on the peak rod burnup limit, is considered in the determination of core average enrichment.

This information is required in order to ensure accuracy and completeness of the technical description contained in the FSAR.

04.03-10

In FSAR Section 4.3.2.2.5, the applicant states that fuel manufacturing practices for modern nuclear fuel have largely eliminated the potential for significant fuel densification and that a power spike factor of 1.0 is used for the U.S. EPR fuel, based on referenced topical report BAW-10163P-A, "Core Operating Limit Methodology for Westinghouse-Designed PWRs." In response to RAI 33 Question 04.03-06, the applicant states that the justification provided in BAW-10163P-A of a power spike factor of 1.0 is applicable to the U.S. EPR due to similarities of the U.S. EPR fuel assembly to a typical U.S. four-loop PWR fuel assembly and due to AREVA's fuel manufacturing practices.

Section 4.1.2.3 of BAW-10163P-A states that the "effects of fuel densification on the peak linear heat rate are included by determining the core average linear heat rate based upon densified fuel" and because "gaps are eliminated or reduced during power operation," and that "no explicit penalty is included to account for densification spike effects." The same section of BAW-10163P-A also states that gap measurements have been performed on irradiated fuel, and only very small gaps have been observed at cold conditions, concluding that the gaps are eliminated or reduced at power conditions. It is noted that the last paragraph of Section 4.1.2.3 states that "the uncertainty factors described in this section are applicable to the Mark BW fuel assembly" and that "appropriate uncertainty factors will be applied to other fuel assembly designs." BAW-10163P-A does not address its potential applicability to other fuel assembly designs.

The applicability of a power spike factor of 1.0 to U.S. EPR fuel requires additional justification, as follows:

a. Describe how the effects of fuel densification on peak linear heat rate are included in the determination of the U.S. EPR core average linear heat rate in lieu of a power spike factor Explain the adequacy of such treatment of local power peaking as an averaged parameter.
b. Provide a comparison of manufactured pellet density (% TD) and fuel rod pre-pressurization between the U.S. EPR fuel and the U.S. PWR fuel evaluated in BAW-10163P-A, "Core Operating Limit Methods for Westinghouse-Designed PWRs" and ANP-10263P-A, "Codes and Methods Applicability Report for the U.S. EPR."

This information is necessary in order to determine whether the effects of fuel densification on local power peaking need to be included in the U.S. EPR nuclear design analysis, in accordance with criteria contained in NUREG-0800 SRP Section 4.3.

04.03-11

The applicant states in FSAR Section 4.3.2.2.7 that F_Q is continuously monitored using the fixed incore SPND detectors, but there is insufficient detail for the staff to complete its evaluation of how F_Q is monitored. There is no Technical Specification limiting condition for operation or surveillance requirement on F_Q. The Technical Specifications Bases Section B 3.2.2 states that F_Q is a direct input parameter to the LOCA analysis; however, the Technical Specification surveillance requirement is on $F^N_{\Delta H}$.

Describe how F_Q is continuously monitored, and explain why there is no Technical Specification surveillance requirement on F_Q. Also, clarify the relationship of F_Q to $F^N_{\Delta H}$ relative to the LOCA analysis.

This information is needed in order to clarify the basis for hot channel factor F_Q and to ensure adequate core monitoring for protection of the core against postulated LOCA events, per the requirements of GDC 13.

04.03-12

In FSAR Tier 2 Section 4.3.2.2.6, the applicant states that $F^N_{\Delta H}$ is used for establishing control rod patterns and control rod bank sequencing, as well as fuel loading patterns. FSAR Tier 2 Section B 3.2 describes the basis for the $F_{\Delta H}$ Technical Specification limiting condition for operation to be also associated with the LOCA analysis.

Provide a complete description of the design basis for hot channel factor $F^N_{\Delta H}$, including its relationship to hot channel factor F_Q, and how it is monitored during power operation.

This information is needed in order to clarify the basis for hot channel factor $F^N_{\Delta H}$ and to thus ensure compliance with GDC 13.

04.03-13

FSAR Tier 2 Section 4.3.2.2.6 states that crud deposition or boron buildup on fuel rods can affect core power distribution and that continuous monitoring of DNBR, LPD, and axial offset would detect changes in power distribution caused by these phenomena.

Provide an explanation of the crud and boron deposition phenomena and describe the effects on core power distribution and core reactivity, how it is accounted for in the design analysis, and how it is detected during reactor operation. This information is required in order to ensure that the requirements of GDC 10 and GDC 12 are met.

04.03-14

FSAR Tier 2 Section 4.3.2.7 states that Technical Specifications require periodic comparison of measured versus calculated power distributions throughout cycle lifetime.

Provide reference to the FSAR Tier 2 Chapter 16 section that requires periodic comparison of measured versus calculated power distributions throughout cycle lifetime.

This information is needed in order to ensure accuracy of the technical description contained in the FSAR.

04.03-15

In FSAR Tier 2 Section 4.3.2.2.7 the applicant states that uncertainties in the aeroball measurement system readings are addressed through comparison of symmetric channels.

Provide a description of how the symmetric aeroball measurements are utilized to address the aeroball measurement system uncertainties. Also, explain how the uncertainties are treated in the determining the measured power distributions and associated fuel thermal parameters such as DNBR, LPD, and hot channel factors.

This information is needed in order to ensure adequate treatment of on-line power distribution measurement uncertainties and hence compliance with GDC 10, GDC 12, and GDC 13.

04.03-16

FSAR Tier 2 Section 4.3.2.2.8 provides a description of the physics tests that are included in initial test program described in FSAR Section 14.2. The staff has evaluated the physics tests described in FSAR Section 14.2 against the Tier 1 FSAR ITAAC requirements and against the guidance contained in Regulatory Guide 1.68, and finds the following testing not included:

a. Determination of core linear power density (LPD) and DNBR in order to verify capability of the core monitoring system for use in complying with Technical Specification limiting conditions for operation,

b. Performance of a pseudo-rod-ejection test to verify calculational models and accident analysis assumptions, and

c. Demonstration of the capability of the incore neutron flux instrumentation to detect rod misalignment equal to or less than the Technical Specifications limits for control rod misalignment. (The staff notes that FSAR Tier 2 Section 16 B 3.1.7 states that detection of a rod misalignment may be done through use of the incore aeroball measurement system.)

Address each of the test items listed above by either adding a test requirement to the initial test program in FSAR Tier 2 Section 14.2, or by justifying not including such test requirement.

The information is needed in order to ensure adequacy of the core physics tests which are part of the initial test program described in FSAR Tier 2 Section 14.2.

04.03-17

In FSAR Tier 2 Section 4.3.2.4, the applicant states that the soluble boron used for reactivity control is natural B^{10} abundance, whereas in FSAR Tier 2 Section 6.8.3 the applicant states that enriched boron is utilized in the extra borating system. FSAR Tier 2 Table 6.8-1 lists the extra boron system B^{10} enrichment to be 37%.

Confirm the use of natural boron for normal (chemical and volume control system) core reactivity control. Also, describe the plant controls for keeping the natural boron separate from the enriched boron, the station procedures for verifying the isotopic level of B^{10} and the plant controls required to ensure the correct boron solutions are used in their respective systems.

This information is required in order to clarify the isotopic boron solutions used for both normal plant operation (chemical and volume control system) and for accident conditions (extra borating system) and to ensure adequate controls are in place to manage the boron isotopic solutions.

04.03-18

FSAR Tier 2 Section 4.3.3.2.4 states that the steady-state NEMO methodology has been benchmarked against the PRISM methodology, and refers to ANP-10286P, "U.S. EPR Rod Ejection Accident Methodology" for comparison of results. The steady-state NEMO methodology was approved by the staff in BAW-10180A, "NEMO-Nodal Expansion Method Optimized;" however, it is not explicitly included in the staff's approval of ANP-10263P-A, "Codes and Methods Applicability Report for the U.S. EPR." Moreover, ANP-10286P, "U.S. EPR Rod Ejection Accident Methodology" has not yet been approved by the staff.

Provide an explanation of the application of the NEMO methodology to the U.S. EPR and its validation basis.

This information is needed in order to clarify which analytical methodologies are utilized in the nuclear design of the U.S. EPR and to ensure that only approved methodologies are utilized.

04.04-21

FSAR Tier 2 Section 4.4 does not address thermal-hydraulic conditions during shutdown or low-power operations, per the guidance contained in SRP Section 4.3.

Provide an analysis of the thermal-hydraulic conditions of the U.S. EPR reactor system during shutdown and low-power operation, including analysis of a rapid boron dilution event during shutdown conditions, in accordance with NUREG-1449.

This information is needed in order to ensure adequate evaluation of the thermal-hydraulic conditions under all reactor operating conditions, including shutdown and low-power operation.

04.04-22

In FSAR Tier 2 Section 4.4.1, the applicant refers to "a small fraction of fuel rods damaged" following a postulated accident. GDC 10 and GDC 12, applicable to FSAR Tier 2 Section 4.4, require that specified acceptable fuel design limits not be exceeded during normal operation and AOOs. Guidelines for acceptable levels of fuel damage following a postulated accident are provided in NUREG-0800 SRP 15.0 in terms of off-site dose relative to 10 CFR 100 criteria, also consistent with the criteria stated in FSAR Tier 2 Chapter 15.

Explain what is meant by “a small fraction of fuel rods damaged” following a postulated accident and how this criterion relates to the applicable regulatory requirements.

This information is needed in order to ensure the correctness of the design criteria and consistency with regulatory requirements.

04.04-23

GDC 10 and GDC 12 specify requirements in terms of the specified acceptable fuel design limits. In FSAR Tier 2 Section 4.4.1, the applicant states that specified acceptable fuel design limits are not exceeded during normal operation and AOOs; however, there is no definition of specified acceptable fuel design limits applicable to the U.S. EPR provided in the Tier 2 FSAR.

Define the specified acceptable fuel design limits applicable to the U.S. EPR.

This information is needed to ensure correctness of the thermal-hydraulic design evaluation provided by the applicant in FSAR Tier 2 Section 4.4.4.3 and compliance with GDC 10 and GDC 12 requirements.

04.04-24

In FSAR Tier 2 Section 4.4.4.1, two critical heat flux correlations, ACH-2 and BWU-N, are described via reference to the applicant’s licensing topical reports ANP-10269P and BAW-10199P-A, respectively. (Note that the former is now approved and should carry the suffix –A.) Although the staff has approved the application of these two correlations as part of the LYNXT methodology to the U.S. EPR fuel in ANP-10263P-A, the applicant states in ANP-10263P-A that the longer length fuel rod of the U.S. EPR fuel assembly will be accounted for in the development of the U.S. EPR CHF correlation.

Confirm the applicability of the ACH-2 and BWU-N critical heart flux correlations to the U.S EPR fuel and explain the requirements, plans, and schedule for development of the U.S. EPR critical heat flux correlation.

This information is necessary to ensure adequacy of the DNBR calculations used to ensure compliance with GDC 10 and GDC 12 requirements.

04.04-25

In FSAR Tier 2 Section 4.4.9.3 the applicant refers to a flow mal-distribution penalty applied to the hot assembly.

Provide a description of the flow mal-distribution penalty, including technical basis, derivation, and application in the calculation of DNBR. Also, address whether the mal-distribution penalty includes the effects of asymmetry of the vessel inlet nozzles.

This information is needed to ensure adequacy of the DNBR calculations used to ensure compliance with GDC 10 and GDC 12 requirements.

04.04-26

The applicant states in FSAR Tier 2 Section 4.4.2.6.4 that hydraulic loads on the fuel assemblies were evaluated, but no description of the analysis or results are presented.

Provide a description of the hydraulic loads analysis for the U.S. EPR fuel and vessel components for normal operating conditions and design basis accident conditions, including the effects of assembly lift.

This information is needed in order to satisfactorily evaluate fuel hold down requirements, as called for in NUREG-0800 SRP 4.4 area of review.

04.04-27

In FSAR Tier 2 Section 4.4.2.9.1, the applicant states that uncertainties due to manufacturing tolerances are included in the heat flux hot channel factor, F_Q used in the COPERNIC code calculation of fuel temperature.

Describe the fuel fabrication uncertainties and tolerances that are accounted for in the COPERNIC calculations, and state specifically how pellet chipping is treated.

This information is needed to ensure adequacy of the analyses used to ensure compliance with GDC 10 and GDC 12 requirements.

04.04-28

[Intentionally deleted]

04.04-29

The material provided by the applicant in FSAR Tier 2 Section 4.4 does not address the effect of crud on DNBR and reactor system pressure drop, as called for in NUREG-0800 SRP 4.4 guidance.

Describe how the thermal-hydraulic design accounts for the effects of crud in the DNBR and core pressure drop calculations throughout the reactor coolant system, including how associated drops in reactor coolant system flow are monitored.

This information is needed to ensure adequate thermal-hydraulic analyses have been performed for the U.S. EPR and that the requirements of GDC 10 and GDC 12 are satisfied.

04.04-30

In FSAR Tier 2 Section 4.4.2.10, the applicant states that the design value of the heat flux hot channel factor, F_Q, does not include a specific allowance for azimuthal tilt.

Provide justification for not including an allowance for azimuthal tilt on the heat flux hot channel factor, F_Q.

This information is needed to ensure that specified acceptable fuel design limits are not exceeded during normal operation and AOOs, as required in GDC 10 and GDC 12.

04.04-31

The plant configuration data for the thermal-hydraulic systems are incorporated by reference to the applicable sections of the Tier 2 FSAR. However, some of the data called for in Regulatory Guide 1.206 are not included.

Provide the additional data called for in Regulatory Guide 1.206, in particular the following items as listed in Section C.I.4.4.3.1 of the regulatory guide:

a. Item (4) total volume of each plant component, including emergency core cooling system (ECCS) components, with sufficient detail to define each part (e.g., downcomer, lower plenum, upper head) of the reactor vessel and steam generator,

b. Item (5) length of the flowpath through each volume,

c. Item (6) height and liquid level of each volume,

d. Item (8) lengths and sizes of all safety injection lines, and

e. Item (10) steady-state pressure and temperature distribution throughout the system.

This information is needed to ensure adequacy of the thermal-hydraulic systems described in the Tier 2 FSAR.

04.04-32

[Question moved to Chapters 14 and 15.]

04.04-33

The applicant's description of the effects of fuel rod bow references two licensing topical reports, BAW-10147P-A and BAW-10186P-A. These two reports are also referenced by the applicant's licensing topical report ANP-10263P-A, "Codes and Methods Applicability to the U.S. EPR." In its discussion of fuel rod bow, ANP-10263P-A states that because the U.S. EPR fuel assembly design is characterized by longer length with two additional spacer grids, the fuel rod bow correlation for the U.S. EPR fuel will be provided in a later submittal.

Considering the apparent need for a U.S. EPR-specific fuel rod bow correlation that has not yet been developed, explain the statements made in FSAR Tier 2 Section 4.4.4.1.5 concerning the application of fuel rod bow penalties on the hot channel factors defined for the U.S. EPR. Also provide additional information on the U.S. EPR correlation, including methodology, application, and development schedule.

This information is needed in order to ensure proper treatment of fuel rod and assembly bow in the thermal-hydraulic analysis of the U.S. EPR fuel.

04.04-34

FSAR Tier 2 Section 4.4.2.9.1 states that the effect of fuel densification is included in the calculation of the total uncertainty associated with fuel and cladding temperatures.

Provide an explanation of the effect of fuel densification in the U.S. EPR fuel and describe how its effects are accounted for in the thermal margin analyses for linear power density and DNBR.

This information is needed in order to ensure proper treatment of fuel densification in the thermal margin analysis of the U.S. EPR fuel.

04.04-35

[Question moved to Chapter 14.]

04.06-3

In Section 4.6.1 of the FSAR, it is stated that the CRDM equipment is designed and qualified to operate in the reactor vessel cavity environment. In the response to RAI 95, question 03.09.04-1a, provides the documentation of the qualification of the CRDM to operate in the reactor vessel cavity environment. However, no statement is available in Subsection 4.6.1 that connects the necessary information to satisfaction of the GDC 4 requirement in other sections. Provide a statement in Section 4.6.1 that connects GDC 4 requirement to the discussion in other DCD subsections. Also, since the ASME code requirements do not apply to the non-pressure boundary components of the CRDM, what effect does the failure of non-pressure boundary component of the CRDM have on the pressure boundary components of the CRDM?

04.06-4

[Question moved to Section 3.5.1.]

04.06-5

[Question moved to Section 3.6.2.]

04.06-6

The control rod drive system (CRDS) according to FSAR Section 4.6.2 is part of the environmental qualification program, and thus designed to operate in extreme conditions.

How has AREVA verified that the maximum design pressure and temperature of the CRDS are not exceeded in the safety analyses where the CRDS is assumed to remain operative, and what are the maximum values? This information is required to assure compliance with GDC 4 regarding the CRDM compatible under its adverse postulated environmental conditions.

04.06-7

[Intentionally deleted]

04.06-8

[Question moved to Chapter 7.]

04.06-9

The mechanical adequacy Section 4.6.3 refers to Section 3.9.4.4 where it is stated that to confirm the mechanical adequacy of the CRDS, a prototype testing program was created. According to Section 3.9.4.4, this program comprises performance tests, stability tests and endurance tests. No reference to the results of this test program is given, and it is not clear from Section 3.9.4.4 how the status and results of this test program support Section 4.6 and other sections of the FSAR. The tests are reported to verify the performance of the equipment under a broad range of conditions. However, the range and conditions are not specified. In response to RAI 95, question 03.09.04-1b, the applicant describes two tests that have been completed for the U.S. EPR design, but does not provide a reference to a report or documentation of the type of tests and test results. In addition, the response to RAI 95, question 03.09.04-1b states that testing is currently underway but does not describe tests that are currently in progress.

What is the status and results of the CRDS prototype testing program and the range of environmental conditions that support the FSAR? This information is necessary to evaluate the performance of the CRDS to ensure an extremely high probability of accomplishing their safety functions in accordance with GDC 29.

04.06-10

In Section 4.6.4, it is written that mechanical overheating of the CRDM causes failure of only one RCCA from inserting into the core by gravity, and the other CRDMs remain functional.

What is the rationale for this statement and the reference to the analysis that supports this conclusion? This information is required to confirm the CRDM is designed to fail into a safe state in accordance with GDC 23 in the event of adverse conditions or environments.

Each CRDM is independent of the other CRDMs in respect to electrical and mechanical failure modes of individual CRDM components. However, under adverse environment conditions are there any individual component failures of the CRDMs that may prevent the RCCA from inserting by gravity. Identify the types of adverse conditions analyzed and the CRDM fail safe response results that satisfy the requirements of GDC 23.