ArevaEPRDCPEm Resource

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Sent:	Wednesday, January 28, 2009 8:27 PM
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Subject:	U.S. EPR Design Certification Application RAI No. 184 (1820), FSAR Ch. 3
Attachments:	RAI_184_EMB1_1820.doc

Attached please find the subject requests for additional information (RAI). A draft of the RAI was provided to you on January 26, 2009, and on January 28, 2009, you informed us that the RAI is clear and no further clarification is needed. As a result, no change is made to the draft RAI. 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
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Request for Additional Information No. 184 (1820), Revision 0

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U. S. EPR Standard Design Certification
AREVA NP Inc.
Docket No. 52-020
SRP Section: 03.09.05 - Reactor Pressure Vessel Internals
Application Section: 3.9.5.1

QUESTIONS for Engineering Mechanics Branch 1 (AP1000/EPR Projects) (EMB1)

03.09.05-12

Design Arrangements

SRP 3.9.5 requires that in-core instrumentation support structures are to be reviewed with the reactor internals. FSAR Tier 2, Table 3.9.5-1, lists some of the instrument support components (level measurement probe columns, guide tubes for instrumentation, level measurement probe upper housing). In addition, in FSAR Tier 2, SubSection 3.9.5.1.3.4, there is a brief description of the level monitoring probe column location and function.

The staff review of the Subsection 3.9.5.1 indicate that the applicant did not provide sufficient information to allow the review of the supporting structures design and their liability to potential adverse flow effects. The FSAR should explicitly state whether these structures and their operating environment are similar to those of other currently operating PWR internals and supporting structures. If this is not the case for some supporting structures, explain the differences and provide appropriate flow-induced vibration analysis for those structures. The applicant is requested to provide more details of the instrumentation supporting structures [e.g. thermocouple, water level sensor, in-core nuclear instrumentation system (ICIS), control and drive rod assembly] as well as the relevant flow-induced vibration analysis for these structures. The staff needs this information to assure conformance with GDC-1 and 4. Revise Section 3.9.5 of the FSAR to include sufficient information about the instrumentation supporting structures and their relevant flow-induced vibration analyses.

03.09.05-13

(Ref. EPR FSAR Section 3.9.5.1.3.1):

The applicant stated in Subsection 3.9.5.1.3.1 of the FSAR that the upper support assembly flange rests on the hold-down spring, which rests on the core barrel flange, which in turn is supported on the ledge machined in the RPV flange. The upper support assembly flange is held in place and preloaded by the RPV closure head flange. Its outer diameter is customized to the corresponding vessel dimension in order to control the radial gap between flanges. The radial gap controls lateral displacements during both normal and postulated accident conditions.

The staff's review of the FSAR indicated that the applicant did not discuss how the horizontal loads on the upper core support assembly due to flow, vibration, seismic and pipe rupture events are transmitted from the upper core support flange to the RV head and hold-down spring by friction or direct contact with the RV flange. It is also not clear if the head and vessel alignment pins also transmit some of the horizontal loads. In addition, the FSAR does not address the potential loss of preload in the hold-down spring due to stress relaxation during service, and the potential effect of preload loss on the functional and structural integrity of upper core support assembly. The applicant is requested to provide an assessment of the potential loss of preload of the hold-down spring due to stress relaxation during the design lifetime, and discuss its effect on the horizontal and vertical restraint of the upper core support and core barrel assemblies. Alternately, provide a reference document where this information is available. The staff needs this evaluation for the above mentioned plant components to assure conformance with GDC-2 and 4. Revise the FSAR to include the requested information.

03.09.05-14

(Ref. EPR FSAR Section 3.9.5.1.3.2):

In FSAR Tier 2, Subsection 3.9.5.1.3.2, the applicant provided a description of the U.S. EPR upper reactor internals upper core plate with holes located opposite the fuel assemblies for core coolant outflow. The upper core plate contains fuel alignment pins at each fuel assembly location that position, align, and restrain the fuel assemblies.

The staff reviewed FSAR Subsection 3.9.5.1.3.2 and FSAR Figures 3.9.5-1 and -4, and found that the applicant did not provide sufficient information to allow the review of the upper core plate design and its interfaces with other reactor internals components. The applicant is requested to provide sufficient details about the design of the upper core plate and its interface with the fuel assemblies, core barrel, upper support columns, and lower guide tubes. Also, explain how these component assemblies are evaluated against possible excitation mechanisms of flow-induced vibration. This information is needed to assure conformance with GDC-1 and 4. Revise Section 3.9.5 of the FSAR to include sufficient information about the design arrangement of the upper core plate and associated internals components including a discussion of the evaluation of the potential adverse effects of flow induced vibration and vortex shedding.

03.09.05-15

The applicant stated in Subsection 3.9.5.1.3.3 of the FSAR that the control rod guide assemblies (CRGAs) consist of guide tubes held together with support plates and tie rods. The guide tube assemblies provide a straight, low-friction channel to insert, withdraw, and drop the control rod drive mechanism (CRDM) drive shafts and attached rod cluster control assemblies (RCCAs). The guide tube assemblies are located inside housings and columns. The housings are attached to the top of the upper support plate (USP) and the columns are attached to the bottom of the USP and also to the upper core plate (UCP). The housings and columns also protect the RCCAs from static and hydraulic loads and other mechanical loads.

The staff's review of Figure 3.9.5-1 of the FSAR Tier 2 indicated that the applicant did not provide sufficient information about the horizontal support plates and tie rods inside the CRGAs. The applicant is requested to provide design details together with relevant flow-induced vibration analysis for these supprt plates. In particular, the applicant is requested to provide drawing/sketches of the control rod guide and to clarify any differences of this design from that of other currently operating PWR reactors and their impact on potential flow excitation mechanisms. This information is needed to assure conformance with GDC-1 and 4. Revise Section 3.9.5 of the FSAR to provide the requested information.

03.09.05-16

The FSAR Tier 2, Subsection 3.9.5.3 states that the allowable design or service limits to be applied to the reactor pressure vessel internals and the effects of service environments, deflection, cycling and fatigue limits are addressed in FSAR Tier 2, section 3.9.3.1. According to FSAR Tier 2, Section 3.9.3.1.1 the effects of the environment on fatigue for Class 1 piping and components are addressed in in FSAR Tier 2, Section 3.12 and Section 3.4 of the AREVA's Topical Report ANP-10264NP, "U.S. EPR Piping Analysis and Pipe Support Design Topical Report." A review of the Section 3.12 and the Topical Report indicate that these two documents do not address effects of fatigue on the reactor pressure vessel internals

The applicant is requested to provide an assessment of the service environments and fatigue on the reactor pressure vessel internals, including potential loss of preload due to irradiation stress relaxation in various threaded fasteners, in particular the guide tube hold-down bolts, guide tube support pins and the flexible sleaves, and the neutron reflector tie-rods, and examine its effect on the structural and functional integrity of the components. Alternately, provide a reference document where this information is available. The requested information will assure conformance with GDC-4. Revise the FSAR to either include the requested information, or provide in FSAR Section 3.9.5.3 a reference where this information is available.

03.09.05-17

In Section 3.9.5.3 of the FSAR, the applicant states, "Evaluation of the adequacy of dynamic analyses under steady-state and operational flow transient conditions, and the proposed program for pre-operational and startup testing of flow-induced vibration and acoustic resonance for RPV internals, is addressed in Section 3.9.2. Evaluation of the adequacy of the structural integrity design of the RPV internals is provided in Section 3.9.3. Section 3.6.3 provides a description of the LBB methodology used to eliminate from the design basis the dynamic effects of the pipe ruptures postulated in Section 3.6.2."

The staff reviewed Section 3.9.3 of the FSAR but did not find where the applicant had provided any details regarding the method used to determine the pressure differential for reactor internal components during different operating conditions or to validate the calculated values. The applicant istherefore requested to provide a description and validation of the method for determining the maximum pressure difference for reactor internals during normal, upset, emergency, and faulted service conditions or alternately, provide a reference document where this information is available. The requested information will assure conformance with GDC-1, -2, and -10. Revise the FSAR to include the requested information.

03.09.05-18

FSAR Tier 2, Subsection 3.9.5.3 by reference to FSAR Section 3.9.3 and Appendix 3C identifies the loading conditions that have been considered in the design of U.S. EPR core internal components and structures.

However, FSAR Subsection 3.9.3 and 3C.4.2.1.2 does not include any analysis of bias errors and random uncertainties. The applicant is therefore requested to provide detailed analysis of expected bias errors and random uncertainties included in predicting the vibration responses of reactor pressure vessel internals and other associated plant systems and components, including the internal components of the steam generators. Discuss and substantiate the contributions to bias errors and random uncertainties for each of the following tasks:

1. Modelling and validation of the forcing functions
2. Modelling and validation of the different computer codes listed in Appendix 3C and used for reactor pressure vessel internals.
3. FE modelling and validation of structural dynamic characteristics
4. Combining the forcing functions and system dynamic characteristics to estimate the dynamic response of structures and components
5. Experimental measurements which are used to validate models and analysis, whether these measurements are performed in-plant or in the laboratory by means of scale model testing.

The Staff needs this information to evaluate the (minimum) margin of safety for the dynamic stress of the reactor pressure vessel internals, and thereby assure conformance with GDC-1 and GDC-4. Revise Subsection 3.9.5 of the FSAR to include analysis of bias errors and random uncertainties.

03.09.05-19

The load or stress component, and displacement limits for the reactor internals that affect the safety and operability of the ASME core support structures are summarized in Table 3.9.3-3 of the FSAR.

However, the FSAR does not give any details how the deformation limits will be determined in the design specification, or provide the technical basis for these deformation limits. As stated in SRP Section 3.9.5, deformation limits for reactor internals should be established by the applicant and presented in the safety analysis report, and the basis for these limits should be included. Also, the stresses for these displacements should not exceed the specified limits. The applicant is requested to provide deformation limits and their technical basis in the appropriate sub-sections of FSAR 3.9.5. Alternately, provide a reference document where this information is available. Review of the requested information regarding the reactor internals design is necessary to assure conformance with GDC-2, 4, and 10. Revise the FSAR to either include the requested information, or provide in FSAR Section 3.9.5 a reference where this information is available.

03.09.05-20

According to the recommendations in Appendix A of the SRP 3.9.5, the applicant is expected to evaluate the dynamic response, stress, and design margin of the internal components of the steam generators. This evaluation is expected to address potential adverse flow effects, such as flow-induced vibrations and acoustic resonances, which are caused by the flow within the steam generator as well as the main steam line flow passing over the main steam line branch connection standpipes for the safety relief valves.

In FSAR Tier 2, Section 3.9.2.4, based on past experience on PWR reactors steam generators, the applicant has concluded that the U.S. EPR steam generator upper internals and flow conditions for which they are subjected to are similar to the existing and currently operating steam generators; therefore, no flow induced vibration analyses are planned for these components. In addition, section 3.9.2.4 contains a brief statement about installing permanent sensors on critical piping systems (e.g. the main steam and feedwater piping systems) that will measure the accelerations in each translation direction during operating life of the plant.

SRP 3.9.5, Appendix A specifically require an applicant for a new reactor design to determine the pressure fluctuations and vibration in the applicable plant systems under flow conditions up to and including the full operating power level. In addition, the size of the steam generators and piping layout and design of the U.S. EPR is not exactly the same as the current fleet of PWRs. Therefore, the applicant is requested to include the details of additional analyses for the flow induced vibration in the steam generator internals, main feedwater and steam piping systems, and standpipes for the safety relief valves in the FSAR section 3.9.2. In addition, details and locations of the permanent sensors on critical piping be included in section 3.9.2 of the FSAR. A brief summary of this additional information should also be included in section 3.9.5 of the FSAR. Reference to the more detailed analysis included in Section 3.9.2 should also be made. The staff needs to review the requested information to assure conformance with GDC-4.

03.09.05-21

(Ref. FSAR Sections 3.9.5.1.1 and 3.9.5.1.2.6):

The reactor coolant flow path for the reactor internals is described in Subsections 3.9.5.1.1, 3.9.5.1.2.4, and 3.9.5.1.2.6 of the FSAR Tier 2. The applicant states that the main coolant flow enters the bottom of the reactor vessel and turns upward, flowing past the flow distribution device and directed through the lower support plate. The lower support plate has inlet holes under each fuel assembly location which have an orifice at their base to equalize the flow rates at the fuel assembly inlets. The flow distribution device provides a homogenous flow distribution between the lower support plate holes

The design and arrangement of the RPV internals controls the distribution of main coolant inlet flow into the fuel assemblies during normal operation which must meet fuel assembly core inlet requirements. However, the FSAR does not provide any details about these flow requirements, or how the internals design and arrangement assure compliance with the core flow requirements, or how the the core flow requirements are verified. The applicant is requested to provide additional details regarding the fuel assembly core inlet requirements, and explain how the detail design of the lower internals assures compliance with the flow requirements, and how the core flow requirements during reactor operation are verified. The requested information is needed to confirm compliance with GDC- 4 and -10 regarding the design of the core support structure to assure that acceptable fuel design limits are not exceeded. Revise the FSAR to include the requested information.

03.09.05-22

The applicant states in FSAR Tier 2, Subsection 3.9.5.1.2.2 that the core barrel outlet nozzle external radius is customized to the corresponding RPV nozzles in order to control the radial gap. The radial gap restricts the bypass flow between the RPV inlet and outlet nozzles. However, the applicant did not assess the potential of the flow induced vibration due to the core barrel flange leakage. The applicant is requested to discuss the potential flow-induced vibration caused by the leakage (or bypass) flow between the outlet nozzle of the core barrel flange and the RV exit nozzle. Since the diameter of core barrel flange is larger than that of currently operating PWR reactors, its shell modes may have lower frequencies, and the leakage flow rate may be higher than currently operating PWR reactors. Provide evidence showing that the leakage flow between the outlet nozzle of the core barrel flange and the RPV exit nozzle will not cause excessive vibration of the core barrel flange. This assessment is needed to assure conformance with GDC- 4. Revise the appropriate sub-section of FSAR Section 3.9.5 to include an assessment of the leakage flow effects on the core barrel flange.