ArevaEPRDCPEm Resource

From:	WELLS Russell (AREVA) [Russell.Wells@areva.com]
Sent:	Wednesday, March 23, 2011 6:25 PM
To:	Tesfaye, Getachew
Cc:	FLECK Sherri (AREVA); COLEMAN Sue (AREVA); BHAGWAGAR Tehemton (AREVA); BENNETT Kathy (AREVA); DELANO Karen (AREVA); HALLINGER Pat (EXTERNAL AREVA); ROMINE Judy (AREVA); RYAN Tom (AREVA); WILLIFORD Dennis (AREVA)
Subject:	DRAFT Response to U.S. EPR Design Certification Application RAI No. 467, FSAR Ch. 3, Questions 03.09.02-155, 156, 162, and 167
Attachments:	RAI 467 Questions 3-9-2-155, 156, 162, 167 Response US EPR DC - (DRAFT).pdf

Getachew

Attached is a draft response to RAI No. 467, Questions 03.09.02-155, 156, 162, and 167 in advance of the final date of April 28, 2011.

Let me know if the staff has questions or if the draft responses can be sent as a final responses.

Sincerely,

Russ Wells
U.S. EPR Design Certification Licensing Manager
AREVA NP, Inc.
3315 Old Forest Road, P.O. Box 10935
Mail Stop OF-57
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Russell.Wells@Areva.com

From: WELLS Russell (RS/NB)
Sent: Thursday, February 24, 2011 2:08 PM
To: 'Tesfaye, Getachew'
Cc: DELANO Karen (RS/NB); ROMINE Judy (RS/NB); BENNETT Kathy (RS/NB); BRYAN Martin (External RS/NB)
Subject: Response to U.S. EPR Design Certification Application RAI No. 467, FSAR Ch. 3

Getachew,

Attached please find AREVA NP Inc.'s response to the subject request for additional information (RAI). The attached file, "RAI 467 Response US EPR DC.pdf" provides a schedule for a technically correct and complete response to the 14 questions.

The following table indicates the respective pages in the response document, "RAI 467 Response US EPR DC.pdf" that contain AREVA NP's response to the subject questions.

Question #	Start Page	End Page
RAI 467 — 03.06.03-28	2	2
RAI 467 — 03.09.02-155	3	3

RAI 467 — 03.09.02-156	4	4
RAI 467 — 03.09.02-157	5	5
RAI 467 — 03.09.02-158	6	6
RAI 467 — 03.09.02-159	7	7
RAI 467 — 03.09.02-160	8	8
RAI 467 — 03.09.02-161	9	9
RAI 467 — 03.09.02-162	10	10
RAI 467 — 03.09.02-163	11	11
RAI 467 — 03.09.02-164	12	12
RAI 467 — 03.09.02-165	13	13
RAI 467 — 03.09.02-166	14	14
RAI 467 — 03.09.02-167	15	15

A complete answer is not provided for the 14 questions. The schedule for a technically correct and complete response to these questions is provided below.

Question #	Response Date
RAI 467 — 03.06.03-28	August 15, 2011
RAI 467 — 03.09.02-155	April 28, 2011
RAI 467 — 03.09.02-156	April 28, 2011
RAI 467 — 03.09.02-157	April 28, 2011
RAI 467 — 03.09.02-158	April 28, 2011
RAI 467 — 03.09.02-159	April 28, 2011
RAI 467 — 03.09.02-160	April 28, 2011
RAI 467 — 03.09.02-161	April 28, 2011
RAI 467 — 03.09.02-162	April 28, 2011
RAI 467 — 03.09.02-163	April 28, 2011
RAI 467 — 03.09.02-164	April 28, 2011
RAI 467 — 03.09.02-165	April 28, 2011
RAI 467 — 03.09.02-166	April 28, 2011
RAI 467 — 03.09.02-167	April 28, 2011

Sincerely,

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From: Tesfaye, Getachew [mailto:Getachew.Tesfaye@nrc.gov]
Sent: Wednesday, January 26, 2011 3:34 PM
To: ZZ-DL-A-USEPR-DL
Cc: Reichelt, Eric; Terao, David; Wong, Yuken; Dixon-Herrity, Jennifer; Miernicki, Michael; Colaccino, Joseph; ArevaEPRDCPEm Resource
Subject: U.S. EPR Design Certification Application RAI No. 467 (5333, 5344), FSAR Ch. 3

Attached please find the subject requests for additional information (RAI). A draft of the RAI was provided to you on January 6, 2011, and discussed with your staff on January 20 and 24, 2011. No change is made to the draft RAI as a result of those discussions. 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,
Getachew Tesfaye
Sr. Project Manager
NRO/DNRL/NARP
(301) 415-3361

Hearing Identifier: AREVA_EPR_DC_RAIs
Email Number: 2744

Mail Envelope Properties (1F1CC1BBDC66B842A46CAC03D6B1CD4104246BA6)

Subject: DRAFT Response to U.S. EPR Design Certification Application RAI No. 467, FSAR Ch. 3, Questions 03.09.02-155, 156, 162, and 167
Sent Date: 3/23/2011 6:24:51 PM
Received Date: 3/23/2011 6:24:54 PM
From: WELLS Russell (AREVA)

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Files	**Size**	**Date & Time**
MESSAGE	4272	3/23/2011 6:24:54 PM
RAI 467 Questions 3-9-2-155, 156, 162, 167 Response US EPR DC - (DRAFT).pdf	453092	

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

Response to

Request for Additional Information No. 467(5333, 5344), Revision 0

**Questions 03.09.02-155, 03.09.02-156, 03.09.02-162,
and 03.09.02-167**

1/26/2011

**U.S. EPR Standard Design Certification
AREVA NP Inc.
Docket No. 52-020
SRP Section: 03.06.03 - Leak-Before-Break Evaluation Procedures
SRP Section: 03.09.02 - Dynamic Testing and Analysis of Systems Structures and Components
Application Section: FSAR Chapter 3**

**QUESTIONS for Component Integrity, Performance, and Testing Branch 1 (AP1000/EPR Projects) (CIB1)
QUESTIONS for Engineering Mechanics Branch 2 (ESBWR/ABWR Projects) (EMB2)**

Question 03.09.02-155:

The NRC Standard Review Plan NUREG-0800, Section 3.9.2.III.5 (page 3.9.2-29), Revision 3 recommends that the staff review the detailed information on whether adequate analysis has been made of the reactor internal structures and unbroken loops to withstand dynamic loads from the most severe LOCA in combination with the SSE. The staff reviewed the information presented in U.S. EPR FSAR Section 3.9.2.5 and Appendix 3C. The staff could not find any description of analytical methods or investigations to specifically determine the stability in compression of reactor vessel internals, particularly the core barrel and other elements of reactor internals, subject to combined SSE and LOCA loads. The applicant is requested to provide description of analytical methods or other investigations to determine stability of the core barrel and other compressive elements of reactor internals.

Response to Question 03.09.02-155:

Core barrel and other core support structures are analyzed to meet the requirements of ASME Boiler and Pressure Vessel (BPV) Code, Section III, Subsection NG-3000. The components are analyzed per the applicable paragraphs of the ASME Code for Design and Service Level Conditions (i.e., NG-3221 through NG-3225). As required by NG-3211(c), for configurations subject to compressive stresses, critical buckling stress is taken into account.

Combined safe shutdown earthquake (SSE) and loss of coolant accident (LOCA) (and other applicable) loads are evaluated as part of Level D conditions. NG-3225 allows the use of ASME BPV Code, Section III, Non-Mandatory Appendix F for evaluating Level D conditions.

Core support structures (i.e., components) subjected to compressive loads are evaluated against buckling limits per F-1331.5 for Level D conditions. Maximum compressive load (or stress) is limited to a value established by:

a) Two-thirds of the value of buckling load (or stress) determined by a comprehensive analysis which considers effects such as geometric imperfections, deformations due to existing loading conditions, nonlinearities, large deformations, residual stresses, and inertial forces;

b) a value equal to 150 percent of the limit established by the rules of NG-3133, except that the pressure difference is permitted to be 250 percent of the given value when the ovality is limited to 1 percent or less

A comprehensive buckling analysis is performed for core support structures subjected to compressive loads using ANSYS finite element software. Appropriate 2-D or 3-D models are created. Material nonlinearities and large deformations are considered. Material nonlinearity is simulated using appropriate plasticity models. The yield strength of the material S_y, is based on the ASME Code. Large deflection command NLGEOM in ANSYS is enabled to account for geometric nonlinearity. The initial geometric imperfections are conservatively simulated based on the tolerances of the assembly.

The full magnitude of the non-dominant loads (e.g., dead weight) is applied at the first load step. The dominant loads are increased incrementally until the solution begins to diverge. At this point, the analysis is stopped and the critical buckling state is considered to have been reached. The loads applied during the load step prior to divergence are considered as the critical buckling loads. The combined SSE and LOCA (and other applicable) loads are favorably compared against two-thirds value of the critical buckling loads to demonstrate acceptance of the design.

Similarly, core support structures that are considered linear supports subjected to compressive loads are evaluated against buckling limits per ASME Code, Section III, F-1334 for Level D conditions.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 03.09.02-156:

The Standard Review Plan, Section 3.9.2.III.5 (page 3.9.2-29) recommends that the staff review the detailed information provided by the applicant on the following subjects:

1. The degree of compliance of the analysis performed with the acceptance criteria listed, and
2. The verification that stresses under the combined loads are within allowable limits of the applicable codes and the deformations are within the limits set to ensure the ability of reactor internal structures to perform the needed safety functions.

The staff reviewed the U.S. EPR FSAR, Section 3.9.2.5 and Appendix 3C. The staff determined that no information was provided to determine the satisfaction of the above SRP criteria. The applicant is requested to provide the necessary information.

Response to Question 03.09.02-156:

NUREG-0800, SRP 3.9.2, Rev. 3 requires a detailed discussion of the reactor internals, design criteria and dynamic analyses methodology for various design and service level conditions. The results of the analyses are required to meet the stress limits of the ASME Code, Section III, Subsection NG for core support structures, and the functional requirements of the reactor internals design specification. Meeting the requirements of the ASME Code, Section III and the design specification provides assurance of the structural and functional integrity of the reactor internals. Additionally, the information requested by NRC will be verified through the following COL information items and inspections, tests, analyses, and acceptance criteria (ITAAC):

- U.S. EPR FSAR Tier 2, Section 3.9.10.2 states: “A COL applicant that references the U.S. EPR design certification will provide a summary of reactor core support structure maximum total stress, deformation, and cumulative usage factor values for each component and each operating condition in conformance with ASME Section III Subsection NG.”

- U.S. EPR FSAR Tier 2, Section 3.9.8 states: “A COL applicant that references the U.S. EPR design certification will prepare the design specifications and design reports for ASME Class 1, 2, and 3 components, piping, supports and core support structures that comply with and are certified to the requirements of Section III of the ASME Code.”

- U.S. EPR FASR Tier 1, Table 2.2.1-5, ITAAC Item 3.16 requires that “RPV internals listed in Table 2.2.1-1 are designed in accordance with ASME Code Section III, Subsection NG.”

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 03.09.02-162:

The applicant provided FSAR Appendix 3C, Figure 3C-8 (page 3C-38), "RPV Isolated Structural Model" to illustrate the modeling of the RPV internals. Physically the core barrel is supported in the spigot of the reactor vessel shell at the core barrel flange. The flange is supported at its bottom surface by the reactor vessel spigot, while on the top surface the core barrel flange is loaded by the hold down spring. Further, the flange of the upper support plate rests on the hold down spring.

The staff determined that the applicant has not provided any details of modeling of the hold down spring, This is critical to determine the upward and rotational (roll and pitch) stability about the two horizontal axis for the core barrel and upper support plate assembly under depressurization pressure loads and SSE acceleration loads.

The applicant is requested to provide details of modeling of the hold down spring, particularly related to the possibilities of core barrel flange and the upper support flange partially (due to rotation about the two horizontal axis) or fully separating from their contacting surfaces, or the spring going solid (reaching full compression), such that the stability of the core barrel and the upper support plate assembly is assured under loads due to LOCA, SSE and other applicable loads.

The applicant is also requested to explain how the dynamic effects of any partial or full separation of the flanges from the contacting surfaces, or the hold down spring going solid, if they occur, are accounted for in the determination of the structural integrity of the core barrel and the upper support assembly.

Response to Question 03.09.02-162:

The AREVA NP structural model of the reactor pressure vessel (RPV) and its internal structures includes springs representing the local stiffnesses of the core barrel flange, upper support plate (USP) flange and Holddown Spring assembly in the vertical direction and in the rotational direction about the horizontal axes (i.e., roll and pitch direction).

The spring rates for the core barrel flange, USP flange and holddown spring were calculated using finite element analyses and are modeled as springs in series. The spring representing the vertical stiffness of the holddown spring incorporates a bilinear stiffness model, in which the stiffness increases rapidly when the holddown spring-to-core barrel and the holddown spring-to-USP gaps are closed due to deflection. However, when these gaps are closed, the overall stiffness of the joint is controlled by the relatively lower stiffnesses of the USP and core barrel flanges, which act in series with the holddown spring.

This connection is preloaded in compression by the installation of the RPV closure head. This compressive force prevents the core barrel flange and USP flange from losing contact with the RPV and RPV closure head contact surfaces.

In the structural model, rotation of the core barrel about the horizontal axes is supported by a rotational spring at the core barrel flange in addition to lateral springs between the core barrel and the RPV at the lower support plate (LSP) elevation. The lateral springs at the LSP provide the primary rotational support based on their distance from this joint and their high stiffnesses.

Rotation of the USP about the horizontal axes is supported solely by the rotational spring at the USP flange. However, there are no significant rotational loads applied to the USP during seismic and pipe rupture events, with the exception of lateral accelerations, that would cause significant rotational deflections.

The dynamic analysis loads prior to and after the hold down spring goes solid are used in the evaluation of the core barrel and the UPS. Full separation of the joint does not occur because of to the preload.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 03.09.02-167:

The staff reviewed FSAR Appendix 3C, and did not find any information if and how the applicant has addressed the following critical and relatively fragile components:

1. Adapter Thermal Sleeves
2. Level Monitoring Probe (LMP)
3. Instrument Guide Tube
4. Rod Cluster Control Assembly (RCCA)
5. Irradiation Specimen Basket.

The applicant is requested to explain how the structural integrity of the above components has been investigated.

Response to Question 03.09.02-167:

U.S. EPR FSAR Tier 2, Table 3.9.5-1 classifies the following components as “Internal Structures” per ASME Boiler and Pressure Vessel Code, Section III, Subsection NG-1122.

- Adapter Thermal Sleeves.
- Instrument Guide Tube.
- Irradiation Specimen Basket.

U.S. EPR FSAR Tier 2, Section 3.9.5.2 states that those components classified as “Internal Structures” are designed to meet the guidelines of NG-3000 and are in accordance with ASME Code Section III, Subsection NG-1122(c) per U.S. EPR FSAR Tier 2, Section 3.9.5.2.

Steady-state and transient loads for various design and service level conditions are applied to the components being designed and resultant stresses are calculated. These stresses are compared with appropriate allowable limits established in NG-3000.

The level measurement probe (LMP) is considered as instrumentation. The pressure retaining components of the LMP are designed to meet the guidelines of NB-3000. The non-pressure retaining structural components of the LMP are designed to meet the guidelines of NG-3000.

U.S. EPR FSAR Tier 2, Section 4.2.16 provides details about the design and analysis of the rod cluster control assemblies (RCCA).

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.