ArevaEPRDCPEm Resource

From: WELLS Russell D (AREVA NP INC) [Russell.Wells@areva.com]
Sent: Wednesday, December 16, 2009 4:09 PM
To: Tesfaye, Getachew
Cc: Pederson Ronda M (AREVA NP INC); BENNETT Kathy A (OFR) (AREVA NP INC); DELANO Karen V (AREVA NP INC)
Subject: Response to U.S. EPR Design Certification Application RAI No. 245, FSAR Ch 3, Supplement 4
Attachments: RAI 245 Supplement 4 Response US EPR DC.pdf

Getachew,

AREVA NP Inc. (AREVA NP) provided responses to 9 of the 22 questions of RAI No. 245 on August 12, 2009. Supplement 1 to AREVA NP's Response to RAI No. 245 provided a response to 1 of the remaining 13 questions, as committed. Revised schedules for the remaining responses were provided in Supplement 2 on December 3, 2009, and Supplement 3 on December 11, 2009. The attached file, "RAI 245 Supplement 4 Response US EPR DC.pdf" provides a technically correct and complete response to the remaining 12 questions, as committed.

Appended to this file are affected pages of the U.S. EPR Final Safety Analysis Report in redline-strikeout format which supports the response to RAI 245 Questions 03.09.02-42 and 03.09.02-45.

The following table indicates the respective pages in the response document, "RAI 245 Supplement 4 Response US EPR DC.pdf" that contain AREVA NP's response to the subject questions.

Question #	Start Page	End Page
RAI 245 — 03.09.02-42 (items 1 and 3)	2	3
RAI 245 — 03.09.02-44	4	4
RAI 245 — 03.09.02-45	5	6
RAI 245 — 03.09.02-46	7	7
RAI 245 — 03.09.02-47	8	10
RAI 245 — 03.09.02-48	11	13
RAI 245 — 03.09.02-49	14	15
RAI 245 — 03.09.02-50	16	17
RAI 245 — 03.09.02-51	18	18
RAI 245 — 03.09.02-53	19	19
RAI 245 — 03.09.02-54	20	20
RAI 245 — 03.09.02-59	21	21

The supporting technical report, ANP-10306P, "Comprehensive Vibration Assessment Program for U.S. EPR™ Reactor Internals Technical Report," was submitted via letter, "Submittal of ANP-10306P, 'Comprehensive Vibration Assessment Program for U.S. EPR™ Reactor Internals Technical Report'," dated December 11, 2009.

Sincerely,

(Russ Wells on behalf of)
Ronda Pederson
ronda.pederson@areva.com
Licensing Manager, U.S. EPR Design Certification
New Plants Deployment
AREVA NP, Inc.
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From: Pederson Ronda M (AREVA NP INC)
Sent: Friday, December 11, 2009 5:58 PM
To: 'Tesfaye, Getachew'
Cc: BENNETT Kathy A (OFR) (AREVA NP INC); DELANO Karen V (AREVA NP INC); HAMMOND Philip R (AREVA NP INC)
Subject: Response to U.S. EPR Design Certification Application RAI No. 245, FSAR Ch 3, Supplement 3

Getachew,

AREVA NP Inc. (AREVA NP) provided responses to 9 of the 22 questions of RAI No. 245 on August 12, 2009. Supplement 1 to AREVA NP's Response to RAI No. 245 provided a response to 1 of the remaining 13 questions, as committed.

Due to word processing and software-related issues experienced while preparing the supporting technical report, ANP-10306P, “Comprehensive Vibration Assessment Program for U.S. EPR™ Reactor Internals Technical Report,” the schedule was revised in Supplement 2 on December 3, 2009.

The referenced report is submitted via letter, “Submittal of ANP-10306P, ‘Comprehensive Vibration Assessment Program for U.S. EPR™ Reactor Internals Technical Report’,” dated December 11, 2009. However, due to administrative processing issues, a revised schedule is provided below for the RAI 245 response document which will submit responses to the remaining 12 questions:

Question #	Response Date
RAI 245 — 03.09.02-42 (items 1 and 3)	December 17, 2009
RAI 245 — 03.09.02-44	December 17, 2009
RAI 245 — 03.09.02-45	December 17, 2009
RAI 245 — 03.09.02-46	December 17, 2009
RAI 245 — 03.09.02-47	December 17, 2009
RAI 245 — 03.09.02-48	December 17, 2009
RAI 245 — 03.09.02-49	December 17, 2009
RAI 245 — 03.09.02-50	December 17, 2009
RAI 245 — 03.09.02-51	December 17, 2009
RAI 245 — 03.09.02-53	December 17, 2009
RAI 245 — 03.09.02-54	December 17, 2009
RAI 245 — 03.09.02-59	December 17, 2009

Sincerely,

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From: Pederson Ronda M (AREVA NP INC)
Sent: Thursday, December 03, 2009 5:48 PM
To: 'Getachew Tesfaye'
Cc: BENNETT Kathy A (OFR) (AREVA NP INC); DELANO Karen V (AREVA NP INC); WELLS Russell D (AREVA NP INC); HAMMOND Philip R (AREVA NP INC)
Subject: Response to U.S. EPR Design Certification Application RAI No. 245, FSAR Ch 3, Supplement 2

Getachew,

AREVA NP Inc. (AREVA NP) provided responses to 9 of the 22 questions of RAI No. 245 on August 12, 2009. Supplement 1 to AREVA NP's Response to RAI No. 245 provided a response to 1 of the remaining 13 questions, as committed.

Due to word processing and software-related issues experienced while preparing the supporting technical report, ANP-10306P, "Comprehensive Vibration Assessment Program for U.S. EPR™ Reactor Internals Technical Report," a revised schedule is provided below for the remaining 12 questions:

Question #	Response Date
RAI 245 — 03.09.02-42 (items 1 and 3)	December 1 , 2009
RAI 245 — 03.09.02-44	December 11, 2009
RAI 245 — 03.09.02-45	December 11, 2009
RAI 245 — 03.09.02-46	December 11, 2009
RAI 245 — 03.09.02-47	December 11, 2009
RAI 245 — 03.09.02-48	December 11, 2009
RAI 245 — 03.09.02-49	December 11, 2009
RAI 245 — 03.09.02-50	December 11, 2009
RAI 245 — 03.09.02-51	December 11, 2009
RAI 245 — 03.09.02-53	December 11, 2009
RAI 245 — 03.09.02-54	December 11, 2009
RAI 245 — 03.09.02-59	December 11, 2009

Sincerely,

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From: WELLS Russell D (AREVA NP INC)
Sent: Thursday, October 29, 2009 1:47 PM
To: 'Getachew Tesfaye'
Cc: Pederson Ronda M (AREVA NP INC); BENNETT Kathy A (OFR) (AREVA NP INC); DELANO Karen V (AREVA NP INC)
Subject: Response to U.S. EPR Design Certification Application RAI No. 245, FSAR Ch 3, Supplement 1

Getachew,

AREVA NP Inc. (AREVA NP) provided responses to 9 of the 22 questions of RAI No. 245 on August 12, 2009. The attached file, "RAI 245 Supplement 1 Response US EPR DC.pdf" provides a technically correct and complete response to 1 of the remaining 13 questions, as committed.

The following table indicates the respective pages in the response document, "RAI 245 Supplement 1 Response US EPR DC.pdf" that contain AREVA NP's response to the subject question.

Question #	Start Page	End Page
RAI 245 — 03.09.04-2	2	3

The schedule for technically correct and complete responses to the remaining 12 questions is unchanged and provided below:

Question #	Response Date
RAI 245 — 03.09.02-42 (items 1 and 3)	December 3, 2009
RAI 245 — 03.09.02-44	December 3, 2009
RAI 245 — 03.09.02-45	December 3, 2009
RAI 245 — 03.09.02-46	December 3, 2009
RAI 245 — 03.09.02-47	December 3, 2009
RAI 245 — 03.09.02-48	December 3, 2009
RAI 245 — 03.09.02-49	December 3, 2009
RAI 245 — 03.09.02-50	December 3, 2009
RAI 245 — 03.09.02-51	December 3, 2009
RAI 245 — 03.09.02-53	December 3, 2009
RAI 245 — 03.09.02-54	December 3, 2009
RAI 245 — 03.09.02-59	December 3, 2009

Sincerely,

(Russ Wells on behalf of)
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From: WELLS Russell D (AREVA NP INC)
Sent: Wednesday, August 12, 2009 4:42 PM
To: 'Getachew Tesfaye'
Cc: Pederson Ronda M (AREVA NP INC); BENNETT Kathy A (OFR) (AREVA NP INC); DELANO Karen V (AREVA NP INC)
Subject: Response to U.S. EPR Design Certification Application RAI No. 245, FSAR Ch 3

Getachew,

Attached please find AREVA NP Inc.'s response to the subject request for additional information (RAI). The attached file, "RAI 245 Response US EPR DC.pdf" provides a technically correct and complete response to 9 of the 22 questions.

Appended to this file are affected pages of the U.S. EPR Final Safety Analysis Report in redline-strikeout format which supports the response to RAI 245 Question 03.09.02-40.

The following table indicates the respective pages in the response document, "RAI 245 Response US EPR DC.pdf," that contain AREVA NP's response to the subject questions.

Question #	Start Page	End Page
RAI 245 — 03.09.02-40	2	3
RAI 245 — 03.09.02-41	4	4
RAI 245 — 03.09.02-42	5	6
RAI 245 — 03.09.02-43	7	11
RAI 245 — 03.09.02-44	12	12
RAI 245 — 03.09.02-45	13	14
RAI 245 — 03.09.02-46	15	15
RAI 245 — 03.09.02-47	16	16
RAI 245 — 03.09.02-48	17	18
RAI 245 — 03.09.02-49	19	19
RAI 245 — 03.09.02-50	20	20
RAI 245 — 03.09.02-51	21	21
RAI 245 — 03.09.02-52	22	22
RAI 245 — 03.09.02-53	23	23
RAI 245 — 03.09.02-54	24	24
RAI 245 — 03.09.02-55	25	25
RAI 245 — 03.09.02-56	26	26
RAI 245 — 03.09.02-57	27	27
RAI 245 — 03.09.02-58	28	28
RAI 245 — 03.09.02-59	29	29
RAI 245 — 03.09.02-60	30	30
RAI 245 — 03.09.04-2	31	31

A complete answer is not provided for 13 of the 22 questions. The schedule for a technically correct and complete response to these questions is provided below.

Question #	Response Date
RAI 245 — 03.09.02-42 (items 1 and 3)	December 3, 2009
RAI 245 — 03.09.02-44	December 3, 2009
RAI 245 — 03.09.02-45	December 3, 2009
RAI 245 — 03.09.02-46	December 3, 2009
RAI 245 — 03.09.02-47	December 3, 2009
RAI 245 — 03.09.02-48	December 3, 2009

RAI 245 — 03.09.02-49	December 3, 2009
RAI 245 — 03.09.02-50	December 3, 2009
RAI 245 — 03.09.02-51	December 3, 2009
RAI 245 — 03.09.02-53	December 3, 2009
RAI 245 — 03.09.02-54	December 3, 2009
RAI 245 — 03.09.02-59	December 3, 2009
RAI 245 — 03.09.04-2	November 13, 2009

Sincerely,

(Russ Wells on behalf of)
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From: Tesfaye, Getachew [mailto:Getachew.Tesfaye@nrc.gov]
Sent: Friday, July 10, 2009 7:36 PM
To: ZZ-DL-A-USEPR-DL
Cc: Spicher, Terri; Dixon-Herrity, Jennifer; Patel, Jay; Miernicki, Michael; Colaccino, Joseph; ArevaEPRDCPEm Resource
Subject: U.S. EPR Design Certification Application RAI No. 245 (2981, 3036),FSAR Ch. 3

Attached please find the subject requests for additional information (RAI). A draft of the RAI was provided to you on June 5, 2009, and discussed with your staff on June 25, 2009. Draft RAI Questions 03.09.02 -53 was modified as a result of that discussion. In addition, the staff has modified Question 03.09.02-49 (shown with yellow highlight) to ensure clarity. 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: 1035

Mail Envelope Properties (1F1CC1BBDC66B842A46CAC03D6B1CD410251C891)

Subject: Response to U.S. EPR Design Certification Application RAI No. 245, FSAR Ch 3, Supplement 4
Sent Date: 12/16/2009 4:08:35 PM
Received Date: 12/16/2009 4:09:07 PM
From: WELLS Russell D (AREVA NP INC)

Created By: Russell.Wells@areva.com

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Tracking Status: None

Post Office: AUSLYNCMX02.adom.ad.corp

Files	**Size**	**Date & Time**
MESSAGE	12330	12/16/2009 4:09:07 PM
RAI 245 Supplement 4 Response US EPR DC.pdf		129160

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

Response to

Request for Additional Information No. 245, Supplement 4

12/03/2009

U.S. EPR Standard Design Certification
AREVA NP Inc.
Docket No. 52-020
SRP Section: 03.09.02 - Dynamic Testing and Analysis of Systems Structures and Components
SRP Section: 03.09.04 - Control Rod Drive Systems
Application Section: 3.9

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

Question 03.09.02-42:

Follow-up to RAI Question 03.09.02-16

In **RAI Question 03.09.02-16**, the staff requested the applicant to provide clarification of how piping attached to the reactor cooling system (RCS) was selected for measurement, the required specifications for the handhelds, and discussion for the plans for their use in characterizing the piping system response relative to the analytical predictions. The applicant responded to **RAI Question 03.09.02-16** in their Response to Request for Additional Information No. 160, Revision 0, by stating that representative piping systems attached to the RCS are monitored by hand held devices and are selected based upon their acoustic connection with the RCS system through acoustic pressure fluctuations. Specifications for hand held devices will be in accordance with the vendor recommendations at the time they are procured. In accordance with the guidance of RG 1.20, Revision 3, the details of the vibration measurement program, including the specifications for the handheld devices, will be included in the comprehensive vibration assessment report which is the responsibility of the COL holder as noted in U.S. EPR FSAR Tier 2, Table 1.8-2. Regarding clarification of how the piping systems are selected, the applicant stated that the representative piping systems are selected based upon their acoustic connection with the RCS system through acoustic pressure fluctuations.

The staff noted that the applicant was also requested to provide the “required specifications” for the handhelds. The “required specifications” refers to requirements that will be used to identify an appropriate device. The “required specifications” for measurement devices is set by the intended use; that is, the environment operated in and the phenomenon that is intended to be measured. The actual device procured may be equal to or better than this “required specification.” Further, to evaluate the appropriateness of the device requirements and their use in this application, the applicant was requested to provide discussion of the plans for their using in characterizing the piping system response relative to the analytical predictions. To propose the use of the device, the applicant must possess a more detailed conception of how the handheld device use will enable adequate description of the vibratory response of the piping systems attached to the RCS. In addition, a reference to a comprehensive vibration program that includes a review of vibration measurement devices for piping was not identified in U.S. EPR FSAR Tier 2, Table 1.8-2. Therefore, the staff is initiating this RAI requesting further clarification of how:

1. piping attached to the reactor cooling system (RCS) was selected for measurement

2. the required specifications for the handhelds

3. discussion for the plans for their use in characterizing the piping system response relative to the analytical predictions.

Response to Question 03.09.02-42:

A response to item 2 of this question was provided in the Response to RAI 245, Question 03.09.02-42, Part 2 on August 12, 2009. Clarifications requested in items 1 and 3 of this question are contained in Comprehensive Vibration Assessment Program (CVAP) Technical Report (AREVA NP Inc. Technical Report ANP-10306P), Appendix A. References to the CVAP Technical Report will be added to the U.S. EPR FSAR Tier 2, Section 3.9.2.1.1 and Table 1.6-1.

FSAR Impact:

U.S. EPR FSAR Tier 2, Section 3.9.2.1.1 and Table 1.6-1 will be revised as described in the response and indicated on the enclosed markup.

Question 03.09.02-44:

Follow-up to RAI Question 03.09.02-17

An additional follow-up to **RAI 03.09.02-17** is required. The applicant stated in the response to **RAI Question 03.09.02-17** that U.S. EPR FSAR Tier 2, Section 3.9.2.4 describes that the RCS, main steam, and main feedwater systems are measured for vibration during initial start-up testing. This section also states the main steam and main feedwater systems will be instrumented with permanent sensors during the operating life of the plant. The staff agrees that if the main steam and main feedwater systems are permanently instrumented and should be capable of identifying acoustic resonances throughout the affected system.

The applicant was requested to discuss how pressure fluctuations would be measured and analyzed to determine loads on any safety related or critical structures. The applicant responded by stating that the details of the vibration measurement, including the use of test results, would be addressed by the COL holder. The staff noted that this position and discussion of the planned pressure instrumentation and the plans for analyzing the pressures to compute loads are not dependent upon the results from the comprehensive RPV vibration assessment program referenced in U.S. EPR FSAR Tier 2, Table 1.8-2, Item 3.9-1. Therefore the staff determined that the applicant is required to provide additional information to complete the review of how pressure fluctuations would be measured and analyzed**.** Therefore, the staff is initiating this RAI requesting further information on the measurement and analysis of pressure fluctuations**.**

Response to Question 03.09.02-44:

During initial startup, strain gauges will be placed on the main steam and feedwater system piping at certain stations to measure pressure oscillations. At each station, strain gauges will be placed around the pipe in a symmetric pattern, and each strain gauge will be oriented circumferentially to provide indication of pressure oscillations inside the piping.

Distances between stations among piping lines in each system will be varied so that, if there is an acoustic resonance, it will not be possible for the distances between stations to have the same acoustic half-wavelength. The two-microphone method will be used to develop the amplitude and phasing of the acoustics, as well as the frequency.

This approach is expected to provide the required sensitivity to indicate acoustic resonance, should that occur. Acoustic resonance would be identified by a high Q factor peak in the power spectral density (PSD) at the frequency of resonance, with high coherence among strain gauge signals at that frequency.

From this information, if acoustic resonance occurs, pressure oscillations observed at components would be computed for evaluation at the appropriate frequency.

FSAR Impact:

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

Question 03.09.02-45:

Follow-up to RAI Question 03.09.02-24

In **RAI Question 03.09.02-24**, the staff requested the applicant to:

a. Provide details of the preoperational vibration and test program which is consistent with the NUREG 0800, SRP Section 3.9.2 subsection II.4 for a prototype. The information requested includes test conditions (e.g. flow conditions, power levels, and temperatures), transducer types, specifications and locations, and methods for preparing the data for comparisons to both the acceptance criteria and the analytical predictions from FSAR Tier 2 Section 3.9.2.3. The applicant is also requested to provide the vibration prediction, test acceptance criteria and bases, and permissible deviations from the criteria prior to the tests. Finally, the applicant should provide a listing of the major reactor internal components that would be subjected to flow induced vibration testing.

b. The applicant has expressed the intent to recategorize the U.S. EPR as a Non-prototype Category I with the Olkiluoto-3 reactor, currently under construction, as the prototype. If the applicant makes this reclassification, per RG 1.20, the applicant is requested to provide the detailed results of the comprehensive vibration assessment program conducted on the Olkiluoto-3 which is consistent with the requirements of RG 1.20 and should include a listing of the major reactor internal components that would be subjected to flow induced vibration testing.

The applicant responded to **RAI Question 03.09.02-24(a)** in their Response to Request for Additional Information No. 160, Revision 0 by stating that, in accordance with the guidance of RG 1.20, Revision 3, details of the preoperational vibration and test program, including the requested information, will be included in the comprehensive vibration assessment program, which is the responsibility of the COL holder as noted in U.S. EPR FSAR Tier 2, Table 1.8-2.

The applicant responded to **RAI Question 03.09.02-24(b)** in their Response to Request for Additional Information No. 160, Revision 0 by stating that in U.S. EPR FSAR Tier 2, Section 3.9.2.4, the U.S. EPR reactor pressure vessel (RPV) internals are classified as prototype design per RG 1.20. Additionally, as stated in U.S. EPR FSAR Tier 2, Section 3.9.2.4, if design changes to the RPV internals are required as a result of the hot functional testing and subsequent inspection at Olkiluoto-3, the appropriate classification of the U.S EPR RPV internals will be determined in accordance with RG 1.20. Accordingly, the associated experimental and/or analytical justification, including any required changes to the comprehensive vibration assessment program, will be provided to the NRC.

The staff noted that the applicant's response to **RAI Question 03.09.02-24(a)** deferred details of the preoperational vibration and test program to the COL holder. The applicant's response to **RAI Question 03.09.02-24(b)** deferred designation of the design as prototype or non-prototype contingent upon the comprehensive vibration assessment program conducted on the as yet unbuilt Olkiluoto-3 plant. However, the requested information in **RAI Question 03.09.02-24(a)** and (b) is inconsistent with what is required and should be available for determining compliance with regulation. The requested information on the vibration assessment program and the prototype design or the justification of classification of the U.S. EPR as non-prototype is needed to complete the DCD review to meet 10 CFR 52.47 to meet 10 CFR 52.47. This requested information has not been provided and therefore this RAI is initiated as a follow-up to request this information.

Response to Question 03.09.02-45:

A) Refer to CVAP Report (AREVA NP Inc. Technical Report ANP-10306P), Section 3.0 through Section 6.0.

B) The first U.S. EPR will have the classification of a “prototype” design.

Reference to the CVAP Report (AREVA NP Inc. Technical Report ANP-10306P) will be added to the U.S. EPR FSAR Tier 2, Section 3.9.2.1.1, Section 3.9.2.3, Section 3.9.2.4, Section 3.9.2.7 and Table 1.6-1. U.S. EPR FSAR Tier 2, Section 3.9.2.3 and Section 3.9.2.4 will be revised to state that the U.S. EPR reactor vessel (RV) internals will be classified “prototype” and to remove references to the Olkiluoto-3 RV internals as being the “prototype” for the U.S. EPR RV internals. U.S. EPR FSAR Tier 2, Table 3.9.2-1 through Table 3.9.2-5 will be deleted.

FSAR Impact:

U.S. EPR FSAR Tier 2, Section 3.9.2.1.1, Section 3.9.2.3, Section 3.9.2.4, Section 3.9.2.7, Table 1.6-1, and Table 3.9.2-1 through Table 3.9.2-5 will be revised as described in the response and indicated on the enclosed markup.

Question 03.09.02-46:

Follow-up to RAI Question 03.09.02-27

In **RAI Question 03.09.02-27**, the staff requested the applicant to provide a discussion of the analyses of these potential adverse flow conditions and the operating conditions that give rise to such flow conditions. The discussion should include the bias errors, uncertainties, and any operational experience the applicant possesses or of which the applicant is cognizant, particularly for situations that have led to past failures, as it relates to the U.S. EPR.

The applicant responded to **RAI Question 03.09.02-27** in their Response to Request for Additional Information No. 160, Revision 0 by stating that in accordance with the guidance of RG 1.20, Revision 3, the details of the assessment of acoustic resonances and self-excited response, along with discussion of the bias errors, uncertainties and operational experience, will be included in the results from the comprehensive vibration assessment program, which is the responsibility of the COL holder as noted in U.S. EPR FSAR Tier 2, Table 1.8-2. The staff requests the applicant to provide the comprehensive vibration assessment program for review by the NRC staff as part of the FSAR to meet 10 CFR 52.47. Therefore, this follow-up RAI is initiated requesting the program for review.

Response to Question 03.09.02-46:

See the Response to RAI 245, Supplement 4, Question 03.09.02-49.

Question 03.09.02-47:

Follow-up to RAI Question 03.09.02-28

In **RAI Question 03.09.02-28**, the staff requested the applicant to supply the results of the analyses so that review of the dynamic properties of the structures and of the methods for obtaining the overall vibration and stress response from the forcing functions, and the vibration and stress models may be made. The results should include:

a. The dynamics of the internal structures, including natural frequencies, mode shapes relevant to the vibration and stress response, damping factors, and the frequency response functions (FRF).

b. The methodology for combining the vibrations and stress response models with the forcing functions to obtain the overall stress and vibration response of the RPV internals.

c. The method for combining the uncertainties and bias errors and the effect of these on the resulting overall stress and vibration response prediction of the RPV internals.

d. The prediction of the overall stress and vibration response for the U.S. EPR RPV internals together with the comparisons to the criteria which demonstrate the stated conformance of the vibration levels with RG 1.20.

The applicant responded to each item in **RAI Question 03.09.02-28** in their Response to Request for Additional Information No. 160, Revision 0 as follows:

a. In accordance with the guidance of RG 1.20, Revision 3, the requested information is addressed in the comprehensive vibration assessment program, which is the responsibility of the COL holder as noted in U.S. EPR FSAR Tier 2, Table 1.8-2. Additionally, the flow-induced vibration (FIV) analyses provide details of the methodology and analysis inputs to the comprehensive vibration assessment program.

b. See Item a above.

c. In accordance with the guidance of RG 1.20, Revision 3, the discussion of the bias errors and uncertainties is part of the results from the comprehensive vibration assessment program. The combined effect of these uncertainties and bias errors on the response of the RPV internals will be assessed after hot functional testing when these inputs are confirmed with test measurements. A comparison of these analysis inputs and their incorporation into the revised prediction of the RPV internals to achieve an agreement between the analytical and test results will be included in the comprehensive vibration assessment program final report.

d. See item a above.

The staff reviewed the applicant's response to **RAI Question 03.09.02-28** and concluded that the applicant needs to provide the comprehensive vibration assessment program for review by the NRC staff as part of the FSAR to meet 10 CFR 52.47. Therefore, this follow-up RAI is initiated requesting a review of the program.

Response to Question 03.09.02-47:

a. The natural frequencies, mode shapes, damping factors, and the response power spectral density (PSD) functions for the full scale reactor vessel (RV) internal components are provided in the Comprehensive Vibration Assessment Program (CVAP) Report (AREVA NP Inc. Technical Report ANP-10306P) as shown in Table 03.09.02-47-1.

b. The methodology for combining the vibrations and stress response models with the forcing functions to obtain the overall stress and vibration response of the RV internals are provided in the CVAP Report (AREVA NP Inc. Technical Report ANP-10306P) as shown in Table 03.09.02-47-2.

c. The methodology for combining uncertainties and bias errors of the turbulent forcing functions and the effect of these on the resulting overall stress and vibration response prediction of the RV internals is provided in the CVAP Report (AREVA NP Inc. Technical Report ANP-10306P) as shown in Table 03.09.02-47-3.

d. The prediction of the overall stress and vibration response for the U.S. EPR RV internals together with the comparisons to the criteria which demonstrate the stated conformance of the vibration levels with RG 1.20 is provided in the CVAP Report (AREVA NP Inc. Technical Report ANP-10306P) as shown in Table 03.09.02-47-4.

Table 03.09.02-47-1—Full Scale RV Internal Component Dynamics

U.S. EPR RV Internals Component	Section of the CVAP Report (ANP-10306P)
RV Lower Internal Assembly	Sections 4.2.5 and 4.2.7
Flow Distribution Device	Sections 4.3.2 and 4.3.4
RV Upper Internals (control room guide assembly (CRGA) column supports, normal column supports, level measurement probe column supports, and guide tubes)	Sections 4.5.1 and 4.5.3
CRGAs and rod cluster control assemblies (RCCA)	Sections 4.6.3 and 4.6.5
Heavy Reflector Tie Rods	Section 4.7.3

Table 03.09.02-47-2—Vibration and Stress Response Combination Methodology

U.S. EPR RV Internals Component	Section of the CVAP Report (ANP-10306P)
RV Lower Internal Assembly	Section 4.2.3.1
Flow Distribution Device	Section 4.3.2
RV Upper Internals (CRGA column supports, normal column supports, level measurement probe column supports, and guide tubes)	Section 4.5.1.1
CRGAs and RCCAs	Section 4.6.3.1
Heavy Reflector Tie Rods	Section 4.7

Table 03.09.02-47-3—Uncertainties and Bias Error Combining Methodology

U.S. EPR RV Internals Component	Section of the CVAP Report (ANP-10306P)
RV Lower Internal Assembly	Section 4.2.2.4
Flow Distribution Device	Section 4.3.2.1
RV Upper Internals (CRGA column supports, normal column supports, level measurement probe column supports, and guide tubes)	Section 4.5.1.1.4
CRGAs and RCCAs	Section 4.6.3.1
Heavy Reflector Tie Rods	Section 4.7

Table 03.09.02-47-4—Stress and Vibration Response

U.S. EPR RV Internals Component	Section of the CVAP Report (ANP-10306P)
RV Lower Internal Assembly	Sections 4.2.6, 4.2.7, 4.2.8 and 5.5
Flow Distribution Device	Sections 4.3.3, 4.3.4, 4.3.5, and 5.5
RV Upper Internals (CRGA column supports, normal column supports, level measurement probe column supports, and guide tubes)	Sections 4.5.2, 4.5.3, 4.5.4, and 5.5
CRGAs and RCCAs	Sections 4.6.4, 4.6.5, 4.6.6, and 5.5
Heavy Reflector Tie Rods	Section 4.7 and 5.5

Reference to the CVAP Report (AREVA NP Inc. Technical Report ANP-10306P) will be added to the U.S. EPR FSAR Tier 2, Section 3.9.2.1.1, Section 3.9.2.3, Section 3.9.2.4, Section 3.9.2.7 and Table 1.6-1.

FSAR Impact:

No additional changes will be made as a result of this question (See FSAR Impact for Question 03.09.02-45).

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Question 03.09.02-48:

Follow-up to RAI Question 03.09.02-29

In **RAI Question 03.09.02-29**, the staff requested the applicant to supply the following information, as recommended by SRP 3.9.2.3 acceptance criteria, that addresses the critical area of flow-excited acoustic and structural resonances or other self-excited response to vortex-induced vibration, turbulence and turbulence buffeting, flow separation, reattachment and impinging flow instabilities:

a. The scale model tests should be discussed with reference to dynamic similarity of the model tests to the full scale structures and operating conditions being analyzed. Additionally, the types and placement of the transducers employed in the small scale model test should be included in the discussion.

b. Because the analysis of the small scale models is used to baseline the analytical/computational procedures for use on the full scale structure, the analytical/computational models of the small scale structures and the analytical procedures employed should be discussed together with an assessment of the bias and uncertainties in the predictions.

c. Comparisons of the small scale model results and the analytical model results should be provided with discussion quality of the comparisons and the implications of the comparison on the use of the procedure on the full scale structure.

d. Discuss the analysis methodologies or software used in the modeling of both the full-scale and the scale model structures. Further, the methodology used to assess the accuracy, limitations and applicability of the software package or analysis procedure should be provided. The discussion of the analysis procedures should include the interaction of the various software packages/models such as providing inputs to each other or any required iterations between models.

e. The applicant stated that "during preoperational testing, the full-scale analytical results are confirmed…." Provide a basis and discussion of the acceptance criteria for confirmation of the results.

f. Because any disagreement between the full scale analysis and the full scale test results will be addressed by adjusting the inputs to the analysis, the identification of the parameters together with the methods and criteria for setting limits on the appropriate adjustment of those input parameters should be provided.

g. The applicant has not specified or referenced locations of transducers or test conditions.

The applicant responded to **RAI Question 03.09.02-29** in their Response to Request for Additional Information No. 160, Revision 0 as follows for each item: The requested information will be provided in the comprehensive vibration assessment program, which is the responsibility of the COL holder as noted in U.S. EPR FSAR Tier 2, Table 1.8-2.

The staff reviewed the applicant's response to **RAI Question 03.09.02-29** and concluded that the applicant needs to provide the comprehensive vibration assessment program for review by the NRC staff as part of the FSAR to meet 10 CFR 52.47. Therefore, this follow-up RAI is initiated requesting a review of the program.

Response to Question 03.09.02-48:

a. The scale model tests that were performed for the reactor vessel (RV) lower internal assembly and the location and type of transducers used during the tests is provided in Comprehensive Vibration Assessment Program (CVAP) Report, Section 4.2.1 (AREVA NP Inc. Technical Report ANP-10306P). A discussion of the methods used to create dynamic similitude in the forcing function between the test and full scale analytical models of the RV lower internal assembly is provided in CVAP Report (AREVA NP Inc. Technical Report ANP-10306P), Section 4.2.2.4 and Section 4.2.5.2.1. The reactor coolant system (RCS) lower internal assembly has been evaluated for the hot standby, the full power operating steady state operating condition, and the 10 percent reactor coolant pump (RCP) overspeed transient condition as reported in CVAP Report (AREVA NP Inc. Technical Report ANP-10306P), Section 4.2.7 and Section 4.2.8.

   Scale model testing to develop the forcing functions of other RV internal components was not undertaken. These forcing functions derive from published data identified in the CVAP Report. See the Response to Question 03.09.02-47, item (c) for the CVAP Report section (AREVA NP Inc. Technical Report ANP-10306P) which addresses the relationship of dynamic similitude with these forcing functions.

b. The analytical/computational procedures used for both the small scale and full scale structure, and the bias and uncertainties associated with the forcing function in the prediction of the response of the RV lower internal assembly is provided in CVAP Report (AREVA NP Inc. Technical Report ANP-10306P), Section 4.2.2 through Section 4.2.5.

c. A comparison of the analytical results obtained with the small scale and full scale models of the RV internal assembly is provided in CVAP Report (AREVA NP Inc. Technical Report ANP-10306P), Section, 4.2.2.3, Section 4.2.3.2 and Section 4.2.4. Information on the degree of comparisons and the implications of the comparison on the use of the procedure on the full scale structure is provided in CVAP Report (AREVA NP Inc. Technical Report ANP-10306P), Section 4.2.8.

d. The analysis methodologies and software used with the modeling of both the scale and full scale model structures is provided in the following sections of the CVAP Report (AREVA NP Inc. Technical Report ANP-10306P).

   Section 4.2.3.1 - For the RV lower internal assembly.

   Section 4.3.2 - For the flow distribution device.

   Section 4.5.1.1 - For the RV upper internals (control rod drive mechanism (CRGA), normal, level measurement probe column supports and the instrumentation guide tubes).

   Section 4.6.3.1 - For the CRGA (tie rods, c-tubes, rod cluster control assembly (RCCA)).

   The methodology used to assess the accuracy, limitations, and applicability of the software package or analysis procedures for the RV lower internal assembly is provided in CVAP Report, Section 4.2.2 (AREVA NP Inc. Technical Report ANP-10306P). For other RV internal components that were evaluated for flow excitation, the AREVA NP Inc. computer codes used are identified in the CVAP Report (AREVA NP Inc. Technical

Report ANP-10306P). Limitations and accuracy verification against the classical closed form solutions are documented in AREVA NP Inc. certification reports and are maintained by appropriate computer software procedures.

e. The acceptance criteria for each RV internal component in regards to the flow induced vibration (FIV) mechanism and hot functional testing is provided in the following sections of the CVAP Report (AREVA NP Inc. Technical Report ANP-10306P).

   Sections 4.2.6 - For the RV lower internal assembly.

   Sections 4.3.3 - For the flow distribution device.

   Sections 4.5.2.3 - For the RV upper internals (CRGA, normal, level measurement probe column supports and the instrumentation guide tubes).

   Sections 4.6.4 - For the CRGA (tie rods, c-tubes, RCCA).

   Information on the basis for confirming the analytical solution against the full scale testing is provided in CVAP Report (AREVA NP Inc. Technical Report ANP-10306P), Section 5.5.

f. See the Response to Question 03.09.02-48, item (e).

g. The location of the transducers on the RV internals and test conditions are provided in CVAP Report (AREVA NP Inc. Technical Report ANP-10306P), Sections 5.2 and 5.3, respectively.

Reference to the CVAP Report (AREVA NP Inc. Technical Report ANP-10306P) will be added to the U.S. EPR FSAR Tier 2, Section 3.9.2.1.1, Section 3.9.2.3, Section 3.9.2.4, Section 3.9.2.7 and Table 1.6-1.

FSAR Impact:

No additional changes will be made as a result of this question (See FSAR Impact for Question 03.09.02-45).

Question 03.09.02-49:

Follow-up to RAI Question 03.09.02-30

The applicant stated in its response to RAI 03.09.02-30 that because transient evaluation of the RPV lower internals to transient conditions will occur during hot functional testing, no analytical evaluation of these transient conditions is planned. The analysis and testing portions of the comprehensive vibration assessment program are intended to compliment, not supplant, each other. The applicant is requested to provide justification for relying solely on the hot functional testing to determine the safety of the plant response to transients and to explain in detail why transient analysis is not performed.

Further, the applicant has stated that if acoustic loadings are observed in the reactor coolant system during hot functional testing, appropriate corrective actions will be taken to eliminate these acoustic loadings. If testing and analysis is conducted only at the full-power, steady-state operating state, as noted above, flow-excited and self-excited response occurring at other flow conditions may be missed. In this follow-up RAI, the applicant is requested to provide the details of their plans to ensure that these conditions are identified and mitigated. Rev 3 of Reg Guide 1.20 states that the applicant should perform a vibration and stress analysis for those steady-state and anticipated transient conditions that correspond to preoperational, initial startup test, and normal operating conditions.

Response to Question 03.09.02-49:

Justification for not performing an analytical evaluation of the reactor vessel (RV) lower internal assembly and the flow distribution device considering the reactor coolant pump (RCP) transients is provided in CVAP Report (AREVA NP Inc. Technical Report ANP-10306P), Section 4.2.8 and Section 4.3.5.

Justification for not performing an analytical evaluation of the RV upper internals considering the RCP transients is provided in CVAP Report (AREVA NP Inc. Technical Report ANP-10306P), Section 4.5.4 and Section 4.6.6.

Regarding the evaluation of structural loading created from sources of acoustic resonance in the RCS, the screening criteria described in the Response to Question 03.09.02-43 and also in CVAP Report (AREVA NP Inc. Technical Report ANP-10306P), Section A.2.1 will be applied with the RCS piping and the piping attached to the RSG considering the operating conditions associated with steady state and all transient conditions to confirm that this source of excitation does not exist in the RCS or re-circulating steam generator (RSG) upper internals. Additionally, the degree of margin or the determination of operating conditions that would lead to acoustic resonance in the RCS piping and attached piping systems will be defined.

As stated in the Response to RAI 245, Question 03.09.02-43, the design of the piping systems attached to the RCS has not been completed. The sensitivities in the arrangement, design, size, and operating conditions on the degree of margin to acoustic resonance will be considered and addressed later in the design process.

Reference to the CVAP Report (AREVA NP Inc. Technical Report ANP-10306P) will be added to the U.S. EPR FSAR Tier 2, Section 3.9.2.1.1, Section 3.9.2.3, Section 3.9.2.4, Section 3.9.2.7 and Table 1.6-1.

FSAR Impact:

No additional changes will be made as a result of this question (See FSAR Impact for Question 03.09.02-45).

Question 03.09.02-50:

Follow-up to RAI Question 03.09.02-31

In **RAI Question 03.09.02-31**, the staff requested the applicant provide a comparison of the U.S. EPR and the German Konvoi plants support columns including the impedances of the mounting arrangements and a comparative analysis or testing that demonstrates the applicability of the German Konvoi experience to the U.S. EPR. The comparison should address placement of the instrumentation and the test conditions intended to evaluate the support columns in the U.S. EPR with those used by the German Konvoi plants.

The staff reviewed the applicant's response to **RAI Question 03.09.02-31** and concurs that the criteria for the FIV analysis of the RPV upper internals has been provided. The acceptance criteria for the random turbulence-induced vibration and for vortex-shedding induced vibrations were acceptable. However, in this follow-up RAI, the applicant is requested to identify the references for the fluid-elastic instability criteria.

Additionally, the applicant was also requested to describe "…any plans for testing to indicate acceptable behavior, including the acceptance criteria, details on the validations of the test plan and the instrumentation and test conditions that will be employed in the U.S. EPR preoperational testing to confirm the acceptable design of the upper internals." The applicant deferred this discussion to the comprehensive vibration assessment program which is the responsibility of the COL holder as noted in U.S. EPR FSAR Tier 2, Table 1.8-2, Item 3.9.1. The staff concludes that the applicant needs to provide the comprehensive vibration assessment program for review by the NRC staff as part of the FSAR to meet 10 CFR 52.47. Therefore, this follow-up RAI is initiated requesting a review of the program in addition to identification of the reference for the fluid-elastic instability criteria.

Response to Question 03.09.02-50:

Comparison of the support column design between the U.S. EPR and the German Konvoi plants is provided in CVAP Report (AREVA NP Inc. Technical Report ANP-10306P), Section 2.3. However, the design of the U.S. EPR support columns is distinct enough that a non-prototype classification is not justified and therefore, an extensive comparison of their design is not provided. Instead, the analytical predictions of the column supports that were performed to various sources of flow excitation are provided in CVAP Report (AREVA NP Inc. Technical Report ANP-10306P), Section 4.5.3.

To assess the margin of safety against fluid-elastic instability, AREVA NP has defined a parameter called fluid-elastic stability margin (FSM), which is the ratio of the critical velocity to the mode shaped weighted pitch velocity:

$$FSM = V_c / V_p$$

The fluid-elastic instability ratio (V_P / V_c), as referred to by some authors, is simply the inverse of the AREVA NP definition for FSM. The computations of the critical velocity and the pitch velocity conform with the requirements of ASME Section III Code, Appendix N-1331.

Reference to the CVAP Report (AREVA NP Inc. Technical Report ANP-10306P) will be added to the U.S. EPR FSAR Tier 2, Section 3.9.2.1.1, Section 3.9.2.3, Section 3.9.2.4, Section 3.9.2.7 and Table 1.6-1.

FSAR Impact:

No additional changes will be made as a result of this question (See FSAR Impact for Question 03.09.02-45).

Question 03.09.02-51:

Follow-up to RAI Question 03.09.02-32

In **RAI Question 03.09.02-32**, the staff requested details of the analyses and testing that indicate acceptable behavior, including the acceptance criteria, details on the validations of the test plan, and the instrumentation and test conditions that will be employed in the U.S. EPR preoperational testing to confirm the acceptable CRGA design.

The applicant responded to **RAI Question 03.09.02-32** in their Response to Request for Additional Information No. 160, Revision 0 by stating that the information is provided in the response to **RAI Question 03.09.02-31**, which stated that the applicant deferred this discussion to the comprehensive vibration assessment program which is the responsibility of the COL holder as noted in U.S. EPR FSAR Tier 2, Table 1.8-2, Item 3.9.1.

Also, in the applicant's response to **RAI Question 03.09.02-32,** they stated in FSAR Tier 2, Section 3.9.2.3 that the full-scale CRGA components have been shown analytically to have acceptable vibrational behavior. This description indicates that the analysis is complete and conclusions indicate that the CRGA design is acceptable. In **RAI Question 03.09.02-32** the applicant was requested to provide details of the analyses. The applicant did not provide this information and the staff initiates this follow-up RAI to request details of the analyses.

Response to Question 03.09.02-51:

The details of the analyses and testing that will indicate acceptable behavior, including the acceptance criteria, details on the validations of the test plan, and the instrumentation and test conditions that will be used in the U.S. EPR preoperational testing, to confirm the acceptable CRGA design, are provided in Sections 4.6 and 5.0 of the CVAP Report (AREVA NP Inc. Technical Report ANP-10306P), Sections 4.6 and 5.0.

References to the CVAP Report (AREVA NP Inc. Technical Report ANP-10306P) will be added to the U.S. EPR FSAR Tier 2, Section 3.9.2.1.1, Section 3.9.2.3, Section 3.9.2.4, Section 3.9.2.7 and Table 1.6-1.

FSAR Impact:

No additional changes will be made as a result of this question (See FSAR Impact for Question 03.09.02-45).

Question 03.09.02-53:

Follow-up to RAI Question 03.09.02-34

In **RAI Question 3.9.34**, the staff requested the applicant to discuss the types of non-destructive testing planned during the inspections process, if walkdowns are included, what monitoring and testing equipment is required, and what actions will be taken as a result of these inspections. It is noted that Tables 3.9.2-3 through 3.9.2-5 reference the storage stands. The applicant should clarify at which points in the testing process components will be removed, placed on storage stands, and inspected.

The staff reviewed the applicant's response to **RAI Question 03.09.02-34** and concluded that the applicant has deferred providing details of the nondestructive testing until after development of the comprehensive vibration assessment program which is the responsibility of the COL holder, as noted in U.S. EPR FSAR Tier 2, Table 1.8-2, Item 3.9.1. The staff concludes that the applicant needs to provide the comprehensive vibration assessment program for review by the NRC staff as part of the FSAR to meet 10 CFR 52.47. Therefore, this follow-up RAI is initiated requesting a review of the program.

Response to Question 03.09.02-53:

The process and type of non-destructive testing planned during the inspections process, the monitoring and testing equipment, and the manner in which the components will be removed from the reactor vessel (RV) and placed on the storage stand is outlined in CVAP Report (AREVA NP Inc. Technical Report ANP-10306P), Section 6.0. As stated in Section 6.0, the inspection results of the RV and the RV internals will be considered acceptable if there is no indication of abnormally large vibration amplitudes or excessive wear.

Reference to the CVAP Report (AREVA NP Inc. Technical Report ANP-10306P) will be added to the U.S. EPR FSAR Tier 2, Section 3.9.2.1.1, Section 3.9.2.3, Section 3.9.2.4, Section 3.9.2.7 and Table 1.6-1.

FSAR Impact:

No additional changes will be made as a result of this question (See FSAR Impact for Question 03.09.02-45).

Question 03.09.02-54:

Follow-up to RAI Question 03.09.02-35

In **RAI Question 03.09.02-35**, the staff requested the applicant to provide a detailed discussion of the basis for the comparison, including acceptance criteria used for determining the relevance of the analytical results and how the results of the analysis using the revised forcing functions are used.

The staff reviewed the applicant's response to **RAI Question 03.09.02-35** and concurs that it is appropriate that results of the hot functional testing and any modifications required to obtain the necessary agreement between the revised analytical solution and the hot functional testing, such as modifications to the forcing function or other analysis inputs, will be included in the comprehensive vibration assessment program final report.

However, prior to the development of the testing program, factors that can influence accurate and meaningful comparison between analytical predictions used in design of the system and test results should be indentified. The applicant was requested to provide a discussion of these factors that influence the comparison of the test results to the analysis and how they will be incorporated into the testing program. The information requested has not been provided and consequently the staff is initiating this follow-up RAI.

Response to Question 03.09.02-54:

The factors influential to the design of the hot functional testing data acquisition system are described in CVAP Report (AREVA NP Inc. Technical Report ANP-10306P), Sections 5.4 and 5.5.

Reference to the CVAP Report (AREVA NP Inc. Technical Report ANP-10306P) will be added to the U.S. EPR FSAR Tier 2, Section 3.9.2.1.1, Section 3.9.2.3, Section 3.9.2.4, Section 3.9.2.7 and Table 1.6-1.

FSAR Impact:

No additional changes will be made as a result of this question (See FSAR Impact for Question 03.09.02-45).

Question 03.09.02-59:

Follow-up to RAI Question 03.09.02-25

In its response to **RAI Question 03.09.02-25**, AREVA provides design information comparing the U.S. EPR steam dryers to those in other similar plants in Table 03.09.02-25-1, "Comparison of U.S. EPR SG Steam Dryers to Other Operating Plants." The NRC staff requests that AREVA provide a comparison of the structural capability of the U.S. EPR steam generator internal parts with those in other similar plants. The staff also requests that AREVA provide a comparison of the U.S. EPR steam system design, layout, branch line size and locations, steam velocity, and other applicable parameters that could affect potential flow-excited and structural resonances in the steam system with those parameters in similar plants. Also, AREVA is requested to include this information in DCD, Tier 2 Section 3.9.2.

Response to Question 03.09.02-59:

In the Response to RAI 245, Question 03.09.02-43 and CVAP Report (AREVA NP Inc. Technical Report ANP-10306P), Appendix A, AREVA has provided clarity on the methods that will be use to screen for sources of acoustic resonances in the piping systems to confirm that this mechanism will not be active. As such, this method will confirm that the excitation of the re-circulating steam generator (RSG) upper internal components, or any other RCS component by this mechanism, is not possible.

Additionally, flow induced vibration (FIV) analyses were performed for steam separators, dryers and other miscellaneous structures in the steam dome considering flow excitation resulting from turbulence and vortex-shedding using upper bound FIV inputs. The results of the analyses concluded that these components are not susceptible to excessive vibrations and are provided in CVAP Report (AREVA NP Inc. Technical Report ANP-10306P), Appendix B and RAI 160, Supplement 1, Question 03.09.02-25 and Question 03.09.02-26.

Therefore, a comparison of the RSG design, its attached piping systems, and the operating conditions between the European and U.S. EPR is not necessary and is not performed.

Reference to the CVAP Report (AREVA NP Inc. Technical Report ANP-10306P) will be added to the U.S. EPR FSAR Tier 2, Section 3.9.2.1.1, Section 3.9.2.3, Section 3.9.2.4, Section 3.9.2.7 and Table 1.6-1.

FSAR Impact:

No additional changes will be made as a result of this question (See FSAR Impact for Question 03.09.02-45).

U.S. EPR Final Safety Analysis Report Markups

Table 1.6-1—Reports Referenced
Sheet 2 of 4

Report No. (See Notes 1, 2, and 3)	Title	Date Submitted to NRC	FSAR Section Number(s)
ANP-10285P ANP-10285NP	U.S. EPR Fuel Assembly Mechanical Design Topical Report	10/02/07	4
ANP-10286P ANP-10286NP	U.S. EPR Rod Ejection Accident Methodology Topical Report	11/20/07	4.3 and 15
ANP-10287P ANP-10287NP	Incore Trip Setpoint and Transient Methodology for U.S. EPR Topical Report	11/27/07	4, 6, 7, and 15
ANP-10288P ANP-10288NP	U.S. EPR Post-LOCA Boron Precipitation and Boron Dilution Technical Report	12/6/07	15
ANP-10290, Revision 1	AREVA NP Environmental Report Standard Design Certification	12/6/07	19.2
ANP-10291P ANP-10291NP	Small Break LOCA and Non-LOCA Sensitivity Studies and Methodology Technical Report	5/09	15
ANP-10292, Revision 1	U.S. EPR Conformance with Standard Review Plan (NUREG-0800) Technical Report	5/09	1.9
ANP-10293	U.S. EPR Design Features to Address GSI-191 Technical Report	2/08	15.6.5.4.3
ANP-10294, Revision 1	U.S. EPR Reactor Coolant Pump Motor Flywheel Structural Analysis Technical Report	3/09	5.4.1.6.6
ANP-10295, Revision 1	U.S. EPR Security Design Features	10/09	13.6
ANP-10296	U.S. EPR Design Features that Enhance Security	12/08	13.6
ANP-10304	U.S. EPR Instrumentation and Control Diversity and Defense in Depth Methodology Technical Report	5/09	7.1, 7.2, 7.3, 7.7, 7.8
ANP-10306P	Comprehensive Vibration Assessment Program for U.S. EPR Reactor Internals Technical Report	12/09	3.9.2.1.1, 3.9.2.3, 3.9.2.4, and 3.9.2.7
BAW-10132-A	Analytical Methods Description – Reactor Coolant System Hydrodynamic Loadings During a Loss-of-Coolant Accident	7/20/79	App. 3C

03.09.02-42 &
03.09.02-45

Preoperational tests are intended to demonstrate that the components comprising these piping systems meet functional design requirements, that piping vibrations are within acceptable levels, and that proper allowance for thermal contraction and expansion is provided. In addition, these tests verify that integrated systems are operating correctly, that system operating procedures are correct, and that system components and safety equipment are operational prior to fuel loading. The end of hot functional testing (HFT) marks the end of Phase I testing. Before fuel loading commences, the results of preoperational tests are evaluated by plant operations and technical staff. If test acceptance criteria are not satisfied, appropriate corrective actions and retesting occurs.

Phase II – Initial Fuel Loading and Pre-Critical Testing

Initial fuel loading and pre-critical tests (refer to Section 14.2.12) are similar to Phase I tests, but occur after the initial reactor core is loaded. Phase II tests establish that the RCS vibration levels and piping reactions to transient conditions (e.g., pump starting and stopping and valve opening and closing) are acceptable. Phase II testing is completed, evaluated, and any required corrective actions taken prior to initiating Phase III (Initial Criticality and Low-Power Physics Testing). If excessive vibration levels are detected during testing, consideration is given to modifying the design specification to re-verify applicable code conformance using the measured vibration as input. If testing and subsequent analysis reveal that additional restraints are needed to reduce stresses to acceptable levels, they are installed.

As described in U.S. EPR Piping Analysis and Pipe Support Design (Reference 2), the U.S. EPR uses snubbers to support piping systems that require free thermal movement but restrained movement due to dynamic loadings. The proper installation and operation of snubbers is verified through visual inspections, hot and cold position measurements, and observation of thermal movements during Phase I and II startup testing. Section 3.9.6 provides the preservice testing (PST) and inservice testing (IST) requirements for snubbers. The IST program incorporates Phase I and II startup testing. Snubber use and locations are determined using the analytical methods presented in Reference 2, as described in Section 3.9.6.

3.9.2.1.1 Piping Vibration Details

03.09.02-42

Piping vibration testing and assessment is performed in accordance with the ASME Standards and Guides for Operation and Maintenance of Nuclear Power Plants (Reference 3) including the addenda. Reference 2 describes the code requirements, acceptance criteria, analysis methods, and modeling techniques for ASME Class 1, 2, and 3 piping and pipe supports. Reference 5 describes the comprehensive vibration assessment program for piping. The Phase I and II tests described above demonstrate that the piping systems withstand vibrations arising from Level A (Normal) loads and Level B (Upset) loads.

3.9.2.2.11 Buried Seismic Category I Piping Conduits, and Tunnels

See Section 3.7.3 and Section 3.10 of Reference 2.

3.9.2.2.12 Interaction of Other Piping with Seismic Category I Piping

See Section 4.4 of Reference 2.

3.9.2.2.13 Analysis Procedure for Damping

See Section 3.7.3 and Section 4.2.5 of Reference 2.

3.9.2.2.14 Test and Analysis Results

See Sections 3.9.2.2.1 and 3.9.2.2.2 above.

3.9.2.3 Dynamic Response Analysis of Reactor Internals Under Operational Flow Transients and Steady-State Conditions

Vibration characteristics and behavior due to flow-induced excitation are complex and not readily determined by analytical means. Thus, the assessment of the vibrational response of the U.S. EPR RPV internals includes a combination of analytical evaluations and testing. Scale model analyses confirmed by scale model tests are used to verify the analytical methods and design inputs that are used for the full-scale design analysis. During preoperational testing, the full-scale analytical results are confirmed, or the analysis inputs are adjusted to achieve agreement and understanding of the response of the RPV internals to flow-induced excitation mechanisms. Reference 5 describes the comprehensive vibration assessment program for reactor internals. The results of this comprehensive vibration assessment program are recorded, consistent with the guidance of RG 1.20.

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The design of the U.S. EPR RPV internals is similar to that of the AREVA NP SAS 4-loop N4 units and the German Konvoï units; however, the U.S. EPR design also incorporates international design evolutions. ~~AREVA NP designates the classification of the RPV internals for the Olkiluoto 3 reactor, which is the first EPR to be constructed, as prototype according to the RG 1.20 classification guidance.~~ ~~T~~Nevertheless, the U.S. EPR internals are classified as prototype per the guidance of RG 1.20. ~~Subsequent to HFT and inspection of the Olkiluoto 3 RPV internals and successfully fulfilling other RG 1.20 requirements for a prototype design, the U.S. EPR internals may be re-classified as non-prototype Category I.~~

Reactor components are excited by flowing coolant, which causes fluctuating pressures on their surfaces. The integration of these pressures over the applied area provides the forcing functions used in the dynamic analysis of the structures. Due to the complexities of the geometries and the random character of the pressure vibrations (FIV). The instrumentation guide tubes are subjected to fuel assembly outlet nozzle turbulence, which is judged to be less than the inlet nozzle turbulence to which incore instrumentation thimbles in previous plants are subjected.

The FIV performance and the design of the flow distribution device and the control rod guide assemblies (CRGAs) have been optimized based on flow tests of these components. The results of these tests are used to optimize the design of these components to minimize vibration levels and the formation of vortices. These test results are also used to optimize the design of the rod control cluster assemblies, which function within the confines of the control rod guide assembly column supports, to minimize wear. Analytical evaluations of these full-scale components have demonstrated that they have acceptable vibrational behavior.

The vibration assessment program demonstrates that the vibration levels of the RPV internals conform to RG 1.20. The U.S. EPR is equipped with a vibration monitoring system (VMS), which provides information on the vibratory behavior of the RPV internals during operation.

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3.9.2.4 Preoperational Flow-Induced Vibration Testing of Reactor Internals

~~Currently, the U.S. EPR RPV internals are classified as a prototype per RG 1.20. AREVA NP plans to rely on the hot functional testing and subsequent inspection of the Olkiluoto 3 internals to reclassify the U.S. EPR internals as non-prototype Category I, provided the other guidelines of RG 1.20 are fulfilled for the Olkiluoto 3 prototype design. The Olkiluoto 3 plant is currently under construction and is due to undergo hot functional testing prior to start of safety-related construction for the U.S. EPR, and qualifies as a valid prototype for the U.S. EPR RPV internals per RG 1.20. If design changes to the RPV internals are required as a result of the hot functional testing and subsequent inspection at Olkiluoto 3, the appropriate classification of the U.S EPR RPV internals will be determined in accordance with RG 1.20.~~

~~Non-prototype Category I designation is applicable to the U.S. EPR RPV internals because the RPV internals for Olkiluoto 3 are a similar design and are subjected to similar flow and thermal hydraulic conditions.~~ The U.S. EPR RPV internals are analyzed for the effects of flow-induced vibration and are not expected to be subject to unacceptable flow induced vibrations. The analysis of the U.S. EPR RPV internals is described in Section 3.9.2.3. ~~Either e~~Extensive measurements ~~or~~and a complete inspection program of the U.S. EPR RPV internals will be performed during hot functional testing in accordance to the guidelines delineated in RG 1.20 for the ~~non-~~ prototype ~~Category I~~ designation. The comprehensive vibration assessment program (Reference 5) outlines the measurement and inspection program. A COL applicant that references the U.S. EPR design certification will submit the results from the vibration assessment program for the U.S. EPR RPV internals, in accordance with RG 1.20.

- Provide reference data to adjust the overall model and for the detailed assessment of the vibratory stresses.
- Provide reference data for the VMS calibration.

The field tests are focused on measuring the vibratory response to the modes likely to be excited by the flow turbulences, fluid-structure interactions, and acoustic sources (i.e., the low frequency modes) for which analytical predictions of the RPV responses have been determined. In addition, the field tests characterize the modes likely to be detected by the VMS.

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Inspections before and after the hot functional test confirm that the RPV internals are functioning correctly. These visual examinations are concerned with the accessible areas of the internals, and in particular the fastening devices, the bearings surfaces, the interfaces between the RPV internal parts that are likely to experience relative motions, and the inside of the RPV. When no indications of harmful vibrations or signs of abnormal wear are detected, and no apparent structural changes are observed, the RPV internals are considered structurally adequate. If such indications are detected, further evaluation is required.

~~The testing and visual inspection plan to be used for the prototype RPV internals at Olkiluoto 3 involves visual inspections before and after the preoperational tests of the internals. These visual examinations are concerned with the accessible areas of the internals, and in particular the fastening devices, the bearings surfaces, the interfaces between the RPV internal parts that are likely to experience relative motions, and the inside of the RPV. Inspections of the lower and upper RPV internals are described in Tables 3.9.2-1, through 3.9.2-5—Visual Inspection of the Inside of the RPV Head While on the Storage Stand.~~

~~The RPV internals flow-induced vibration measurement program is conducted during preoperational tests of the Olkiluoto 3 and U.S. EPR reactors. The U.S. EPR RPV internals testing and inspection programs conform to RG 1.20.~~

RG 1.20, Revision 3, recommends that the potential adverse effects from pressure fluctuations and vibrations in piping systems associated with acoustic resonance should be considered for the steam generator (SG) internals for both PWRs and BWRs. The U.S. EPR SG upper internals (e.g., steam dryers, separators) are subject to secondary side steam flow. Although there are instances of these components in BWR plant designs experiencing excessive vibration resulting from plant power uprate, to date none have been reported for PWR SG designs both internationally or within the United States. This is further supported by a review of the INPO steam generator operating experience database which also does not have any events related to vibration problems for PWR SG upper internals. In response to public comments on the proposed revisions to RG 1.20 (i.e., DG-1163), the NRC states: "In addition to BWR plants, the pressurized-water reactor (PWR) at the Palo Verde plant experienced degradation from excess vibration that had characteristics similar to those of the phenomenon affecting the BWR plants." However, AREVA NP understands that the excessive vibrations associated with the shut down coolant pipe at the Palo Verde plant did not lead to vibration problems with the SG upper internals.

The design of the U.S. EPR SG upper internals and the flow conditions for which they are subjected are similar to the existing and currently operating SGs in the United States and Europe. Based on operational experience and analysis of the SG upper internals, AREVA NP concludes that these non-safety-related components will not experience excessive vibration. Therefore, no flow-induced vibration analyses or startup testing is currently planned for these components. Reference 5 describes the vibration evaluations performed for the U.S. EPR SG upper internals and operating experience with similar SG upper internal designs.

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The vibration of representative trains of piping attached to the RCS as well as main steam and main feedwater lines are measured during initial startup testing. These measurements will be taken at discrete piping locations and also at the other key components (e.g., valves and pumps) installed along the length of pipe. Accelerations will be measured using hand-held devices for both steady-state and transient flow conditions.

In the 1970s and 1980s, the above process was employed to address vibration concerns during startup testing for the current operating plants. Since there are very few instances of excessive pipe vibration while operating these plants at their design power level, this is a proven and reliable method of validation. The majority of the cases in which excessive pipe vibration have been observed have been at the stand off branch lines in the main steam piping system when the plant uprates to a higher power level. Therefore, greater scrutiny of these piping configurations will be stressed during the initial startup testing using the method outlined above. Under certain conditions, some of the stand off branch lines may be instrumented with permanent sensors to monitor their accelerations during the life of the plant.

Excessive vibration or instabilities in piping systems can be difficult to analytically predict for most flow-induced vibration mechanisms. This is due to subtle differences in the thermal hydraulic conditions in the piping that can have a significant effect on the vibration response of these piping components. For these reasons, the critical piping systems (e.g., the main steam and feedwater piping systems) will be instrumented with permanent sensors that will measure the accelerations in each translational direction during the operating life of the plant. The acceptance criteria for the piping and other key components installed along the length of the pipe will be based upon satisfying the appropriate displacement, acceleration, stress, and fatigue limits.

3.9.2.4.1 Exceptions to Regulatory Guide 1.20

The U.S. EPR conforms to Regulatory Guide 1.20 in regards to the reactor internals, piping and components in the reactor coolant, main steam, condensate, and feedwater systems with the following exceptions:

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1. Vibration and Stress Analysis Program and Vibration and Stress Measurement Program requirements of RG 1.20 with regards to steam generator internals as discussed above. Reference 5 provides the justification for their exception to RG 1.20.

2. Vibration and Stress Measurement Program requirements of RG 1.20 with regards to instrumenting the condensate system - Accelerations at discreet locations in condensate system of the U.S. EPR will be measured during start-up testing using hand-held vibration monitors. Hand held vibration measurement equipment is useful as an indicator of piping vibration, in that the personnel using such equipment can identify the locations of high vibration by moving along the piping. By doing so, effective estimates of the severity of piping vibration may be accomplished. Hand held vibration instrumentation is suitable for certain frequency ranges, and the frequency range of the actual instruments used will be identified as part of the test program. This information will be used in the vibration evaluations.

3.9.2.5 Dynamic System Analysis of the Reactor Internals Under Faulted Conditions

The dynamic model used in the analysis of the RPV internals, the RPV Isolated Model, includes the core, the RPV upper and lower internals, the RPV pressure boundary, the reactor coolant loops (piping and components), the RCS supports, and the Reactor Building internal structure. The dynamic analyses consider the effects of the gaps that exist between the vessel and the core barrel, between the vessel and the upper support assembly, between the vessel and the lower support plate, between fuel assemblies, and between the fuel assemblies and the heavy reflector. See Appendix 3C for a representative diagram of the RPV Isolated Model and additional information regarding the dynamic loading analysis of this model.

Analysis of the RPV internals for blowdown loads resulting from a guillotine break of the safety injection line nozzles on the hot and cold legs is performed using direct step-by-step integration methods. Note that breaks are not considered in the main coolant loop piping (hot and cold legs), pressurizer surge line, and main steam line piping (from the steam generators to the first anchor point location) due to the application of leak-before-break methodology to these lines (see Section 3.6.3). The forcing functions obtained from hydraulic analysis of the safety injection line breaks are defined at points in the RPV internals where changes in cross-section or direction of flow occur, such that differential loads are generated during the blowdown transient. Additional details of the structural analysis of the RPV Isolated Model for LOCA loading are given in Appendix 3C.

Analysis of the RPV internals for safe shutdown earthquake (SSE) loading uses direct step-by-step time-history analysis techniques. The SSE analysis of the RPV Isolated Model is described in Appendix 3C.

The response of the RPV internals to SSE loading are combined with their response to the safety injection line breaks by the square-root-of-the-sum-of-the-squares method. Section 3.9.3 provides the faulted load combinations considered in the stress and fatigue analyses of the RPV internals.

3.9.2.6 Correlations of Reactor Internals Vibration Tests with the Analytical Results

The results of the dynamic analysis of the RPV internals are compared to the results of preoperational tests, and this comparison verifies that the analytical model provides appropriate results. If the predicted responses differ significantly from the measured values, the vibration responses are determined with the measured forcing function as input.

3.9.2.7 References

1. ASME Boiler and Pressure Vessel Code, Section III, "Rules for Construction of Nuclear Power Plant Components," The American Society of Mechanical Engineers, 2004.

2. ANP-10264NP-A, Revision 0, "U.S. EPR Piping Analysis and Pipe Support Design Topical Report," AREVA NP Inc., November 2008.

3. ASME OM-S/G-2000, "Standards and Guides for Operation and Maintenance of Nuclear Power Plants," The American Society of Mechanical Engineers, 2000.

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4. Deleted.

5. ANP-10306P, Revision 0, "Comprehensive Vibration Assessment Program for U.S. EPR Reactor Internals Technical Report," AREVA NP Inc., December 2009.

Table 3.9.2-1—Table DeletedVisual Inspection of the Accessible Areas of the Upper Internals While on the Storage Stand

Component	Subcomponent	Inspection
Top surfaces of the upper support assembly	Control rod guide assembly (CRGA) housing	Presence and condition of the bolts and their locking cups
	Level measurement probe (LMP) thimble upper housing	Presence and condition of the bolts and their locking cups
	Head and vessel alignment pins	Presence and condition of the bolts and their locking bars
	Flange	Aspect of the bearing area
Bottom surfaces of the upper support assembly	Normal columns, LMP columns, and accessible CRGA columns	Presence and condition of the bolts and their locking cups
	Flange	Hold down spring contact area
CRGA columns	Accessible guide tubes for instrumentation lance finger	Bracket fastening; presence and condition of the bolts and their spot welds
Upper core plate (UCP) top surface	CRGA columns	Aspect of the flange / CRGA pin interface
	UCP guide pin inserts	Presence and condition of the bolts and their locking cups; aspect of the wear resistant alloy surfaces
UCP bottom surface	Normal columns and LMP columns	Presence and condition of the bolts and their locking bars
	CRGA pins	Presence and condition of the locking device
	Guide tubes for instrumentation lance finger	Bracket fasteners inside the UCP; presence and condition of the bolts and their spot welds
	Upper fuel pins	Presence and condition of the pins and their spot welds

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Table 3.9.2-2—Table DeletedVisual Inspection of the Inside of the Lower Internals in the Reactor Pressure Vessel

Component	Subcomponent	Inspection
Core barrel flange top surface	Head and vessel alignment pins	Presence and condition of the bolts and their locking bars
	Hold down spring contact area	Surface aspect
Heavy reflector top	Upper core plate (UCP) guide pins	Presence and condition of the bolts and their locking cups; aspect of the wear resistant alloy surfaces
	Tie rods	Presence and condition of the nuts and their locking devices
Lower support plate	Access plug fasteners	Presence and condition of the bolts and their locking bars
	Lower fuel pins	Presence and condition of the pins and their spot welds

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Table 3.9.2-3—~~Visual Inspection of the Outside of the Lower Internals While on the Storage Stand~~Table Deleted

~~Component~~	~~Subcomponent~~	~~Inspection~~
~~Irradiation baskets~~	~~Fasteners~~	~~Presence and condition of the bolts and their locking bars~~
~~Radial key inserts~~	~~Insert fasteners~~	~~Presence and condition of the bolts and their locking bars~~
	~~Wear resistant alloy surfaces of the inserts~~	~~Surface aspect~~

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Table 3.9.2-4—Table DeletedVisual Inspection of the Inside of the RPV While on the Storage Stand

Component	Subcomponent	Inspection
Reactor pressure vessel (RPV) flange	Contact surface with the lower internal flange	Surface aspect
Outlet nozzles	Potential contact surface with the lower internal nozzles	Surface aspect
Radial keys	Insert fasteners	Presence and condition of the bolts and their locking bars
	Wear resistant alloy surfaces of the inserts	Surface aspect

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Table 3.9.2-5—Table DeletedVisual Inspection of the Inside of the RPV Head While on the Storage Stand

Component	Subcomponent	Inspection
Reactor pressure vessel (RPV) head flange	Contact surface with the upper internal flange	Surface aspect
Adaptors	Thermal sleeves	Amplitude of vertical displacement

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