AREVA

July 29, 2011
NRC:11:080

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Response to U.S. EPR Design Certification Application RAI No. 422, Supplement 26

In Reference 1, the NRC provided a request for additional information (RAI) regarding the U.S. EPR design certification application. Reference 2 provided a schedule for technically correct and complete responses to RAI No. 422. Reference 3 provided a technically correct and complete response to 2 (Questions 03.09.02-86 and 03.09.02-143) of the 63 questions. Reference 4 provided a revised schedule for a technically correct and complete response to the remaining 61 questions based on additional evaluations. Reference 5 provided a revised schedule for a technically correct and complete response to 6 of the remaining 61 questions based on additional evaluations. Reference 6 provided a revised schedule for a technically correct and complete response to 16 of the remaining 61 questions to allow additional time for AREVA NP to interact with the NRC. Reference 7 provided a technically correct and complete response to 11 (Questions 03.09.02-125, 03.09.02-128, 03.09.02-129, 03.09.02-130, 03.09.02-132, 03.09.02-133, 03.09.02-135, 03.09.02-136, 03.09.02-137, 03.09.02-139, and 03.09.02-141) of the remaining 61 questions. Reference 8 provided a revised schedule for a technically correct and complete response to 13 of the remaining 50 questions to allow additional time for AREVA NP to interact with the NRC. Reference 9 provided a revised schedule for a technically correct and complete response to 5 of the remaining 50 questions to allow additional time for AREVA NP to address NRC comments. Reference 10 provided a revised schedule for a technically correct and complete response to 3 of the remaining 50 questions based on additional evaluations. Reference 11 provided a revised schedule for a technically correct and complete response to 13 of the remaining 50 questions to allow additional time for AREVA NP to interact with the NRC. Reference 12 provided a revised schedule for a technically correct and complete response to 13 of the remaining 50 questions based on additional evaluations. Reference 13 provided a revised schedule for a technically correct and complete response to 16 of the remaining 50 questions based on additional evaluations. Reference 14 provided a revised schedule for a technically correct and complete response to 18 of the remaining 50 questions to allow additional time for AREVA NP to interact with the NRC. Reference 15 provided a technically correct and complete response to 1 (Question 03.09.02-126) of the remaining 50 questions. Reference 16 provided a technically correct and complete response to 8 (Questions 03.09.02-89, -91, -92, -93, -95, -96, -97, 142) of the remaining 49 questions. Reference 17 provided a revised schedule for a technically correct and complete response to 32 of the remaining 41 questions to allow additional time for AREVA NP to interact with the NRC. Reference 18 provided a revised schedule for a technically correct and complete response to 6 of the remaining 41 questions to allow additional time for AREVA NP to interact with the NRC. Reference 19 provided a revised schedule for a technically correct and complete response to 3 of the remaining 41 questions to allow additional time for AREVA NP to interact with the NRC. Reference 20 provided a revised schedule for a technically correct and complete final response to 29 of the remaining 41 questions to allow additional time for AREVA NP to

D077
NRO

interact with the NRC. Reference 21 provided a technically correct and complete response to 3 (Questions 03.09.02-105, -106, -123) of the remaining 41 questions. Reference 22 provided a revised schedule for a technically correct and complete final response to 9 of the remaining 38 questions. Reference 23 provided a technically correct and complete final response to 5 of the 38 remaining questions (i.e., Questions 03.09.02-90, 03.09.02-94, 03.09.02-124, 03.09.02-138, 03.09.02-144). Reference 24 provided a revised schedule for 28 of the remaining 33 questions. Reference 25 provided a revised response to Question 03.09.02-139 that was originally provided to the NRC in Reference 7. Reference 25 provided a technically correct and complete final response to 16 of the 33 remaining questions for RAI 422 and a revised response to Question 03.09.02-136. Reference 26 provided a schedule for technically correct and complete final responses to the remaining 17 questions.

Enclosed is a technically correct and complete final response to 3 of the 17 remaining questions of RAI 422, as shown in the below table. Additionally, the enclosed response contains a revised response to Question 03.09.02-88 which was originally submitted to NRC in Reference 26. The final response that was submitted to Question 03.09.02-88 in Reference 26 inadvertently omitted Figure 03.09.02-88-1 from the RAI response. The enclosed revised response contains this figure.

AREVA NP considers some of the material contained in the attached response to be proprietary. As required by 10 CFR 2.390(b), an affidavit is attached to support the withholding of the information from public disclosure. Proprietary and non-proprietary versions of the enclosure to this letter are provided.

Appended to this file are affected pages of ANP-10306P, "Comprehensive Vibration Assessment Program for U.S. EPR Reactor Internals Technical Report," in redline-strikeout format which support the response to RAI 422 Questions 03.09.02-131 and 03.09.02-140.

The following table indicates the respective pages in the enclosed response that contain AREVA NP's final response to the subject questions.

Question #	Start Page	End Page
RAI 422 — 03.09.02-88	2	4
RAI 422 — 03.09.02-131	5	5
RAI 422 — 03.09.02-134	6	6
RAI 422 — 03.09.02-140	7	9

The schedule for the technically correct and complete final response to the remaining 14 questions is unchanged and is provided below.

Question #	Response Date
RAI 422 — 03.09.02-84	September 13, 2011
RAI 422 — 03.09.02-85	September 13, 2011
RAI 422 — 03.09.02-87	September 13, 2011
RAI 422 — 03.09.02-99	September 13, 2011
RAI 422 — 03.09.02-100	September 13, 2011
RAI 422 — 03.09.02-101	September 13, 2011
RAI 422 — 03.09.02-103	September 13, 2011

Question #	Response Date
RAI 422 — 03.09.02-104	September 13, 2011
RAI 422 — 03.09.02-107	September 13, 2011
RAI 422 — 03.09.02-108	September 13, 2011
RAI 422 — 03.09.02-110	September 13, 2011
RAI 422 — 03.09.02-112	September 13, 2011
RAI 422 — 03.09.02-114	September 13, 2011
RAI 422 — 03.09.02-119	September 13, 2011

If you have any questions related to this submittal, please contact me by telephone at 434-832-2369 or by e-mail to sandra.sloan@areva.com.

Sincerely,

Sandra M. Sloan

Sandra M. Sloan, Manager
New Plants Regulatory Affairs
AREVA NP Inc.

Enclosures

cc: G. Tesfaye
Docket No. 52-020

References

Ref. 1: E-mail, Getachew Tesfaye (NRC) to Martin C. Bryan (AREVA NP Inc.), "U.S. EPR Design Certification Application RAI No. 422 (4792), FSAR Ch. 3," August 3, 2010.

Ref. 2: E-mail, Martin C. Bryan (AREVA NP Inc.) to Getachew Tesfaye (NRC), "Response to U.S. EPR Design Certification Application RAI No. 422, FSAR Ch. 3," September 2, 2010.

Ref. 3: E-mail, Martin C. Bryan (AREVA NP Inc.) to Getachew Tesfaye (NRC), "Response to U.S. EPR Design Certification Application RAI No. 422, FSAR Ch. 3, Supplement 1," September 9, 2010.

Ref. 4: E-mail, Martin C. Bryan (AREVA NP Inc.) to Getachew Tesfaye (NRC), "Response to U.S. EPR Design Certification Application RAI No. 422, FSAR Ch. 3, Supplement 2," September 27, 2010.

Ref. 5: E-mail, Martin C. Bryan (AREVA NP Inc.) to Getachew Tesfaye (NRC), "Response to U.S. EPR Design Certification Application RAI No. 422, FSAR Ch. 3, Supplement 3," November 2, 2010.

Ref. 6: E-mail, Martin C. Bryan (AREVA NP Inc.) to Getachew Tesfaye (NRC), "Response to U.S. EPR Design Certification Application RAI No. 422, FSAR Ch. 3, Supplement 4," November 22, 2010.

Ref. 7: E-mail, Martin C. Bryan (AREVA NP Inc.) to Getachew Tesfaye (NRC), "Response to U.S. EPR Design Certification Application RAI No. 422, FSAR Ch. 3, Supplement 5," December 22, 2010.

Ref. 8: E-mail, Martin C. Bryan (AREVA NP Inc.) to Getachew Tesfaye (NRC), "Response to U.S. EPR Design Certification Application RAI No. 422, FSAR Ch. 3, Supplement 6," January 12, 2011.

Ref. 9: E-mail, Martin C. Bryan (AREVA NP Inc.) to Getachew Tesfaye (NRC), "Response to U.S. EPR Design Certification Application RAI No. 422, FSAR Ch. 3, Supplement 7," January 13, 2011.

Ref. 10: E-mail, Martin C. Bryan (AREVA NP Inc.) to Getachew Tesfaye (NRC), "Response to U.S. EPR Design Certification Application RAI No. 422, FSAR Ch. 3, Supplement 8," January 27, 2011.

Ref. 11: E-mail, Martin C. Bryan (AREVA NP Inc.) to Getachew Tesfaye (NRC), "Response to U.S. EPR Design Certification Application RAI No. 422, FSAR Ch. 3, Supplement 9," February 8, 2011.

Ref. 12: E-mail, Martin C. Bryan (AREVA NP Inc.) to Getachew Tesfaye (NRC), "Response to U.S. EPR Design Certification Application RAI No. 422, FSAR Ch. 3, Supplement 10," February 10, 2011.

Ref. 13: E-mail, Martin C. Bryan (AREVA NP Inc.) to Getachew Tesfaye (NRC), "Response to U.S. EPR Design Certification Application RAI No. 422, FSAR Ch. 3, Supplement 11," February 17, 2011.

Ref. 14: E-mail, Russell Wells (AREVA NP Inc.) to Getachew Tesfaye (NRC), "Response to U.S. EPR Design Certification Application RAI No. 422, FSAR Ch. 3, Supplement 12," March 16, 2011.

Ref. 15: E-mail, Russell Wells (AREVA NP Inc.) to Getachew Tesfaye (NRC), "PROPRIETARY Response to U.S. EPR Design Certification Application RAI No. 422, FSAR Ch. 3, Supplement 13," March 24, 2011.

Ref. 16: E-mail, Russell Wells (AREVA NP Inc.) to Getachew Tesfaye (NRC), "Response to U.S. EPR Design Certification Application RAI No. 422, FSAR Ch. 3, Supplement 14," March 30, 2011.

Ref. 17: E-mail, Russell Wells (AREVA NP Inc.) to Getachew Tesfaye (NRC), "Response to U.S. EPR Design Certification Application RAI No. 422, FSAR Ch. 3, Supplement 15," March 31, 2011.

Ref. 18: E-mail, Russell Wells (AREVA NP Inc.) to Getachew Tesfaye (NRC), "Response to U.S. EPR Design Certification Application RAI No. 422, FSAR Ch. 3, Supplement 16," April 13, 2011.

Ref. 19: E-mail, Russell Wells (AREVA NP Inc.) to Getachew Tesfaye (NRC), "Response to U.S. EPR Design Certification Application RAI No. 422, FSAR Ch. 3, Supplement 17," April 20, 2011.

Ref. 20: E-mail, Russell Wells (AREVA NP Inc.) to Getachew Tesfaye (NRC), "Response to U.S. EPR Design Certification Application RAI No. 422, FSAR Ch. 3, Supplement 18," May 11, 2011.

Ref. 21: E-mail, Dennis Williford (AREVA NP Inc.) to Getachew Tesfaye (NRC), "Response to U.S. EPR Design Certification Application RAI No. 422, FSAR Ch. 3, Supplement 19," May 20, 2011.

Ref. 22: E-mail, Dennis Williford (AREVA NP Inc.) to Getachew Tesfaye (NRC), "Response to U.S. EPR Design Certification Application RAI No. 422, FSAR Ch. 3, Supplement 20," May 25, 2011.

Ref. 23: E-mail, Dennis Williford (AREVA NP Inc.) to Getachew Tesfaye (NRC), "Response to U.S. EPR Design Certification Application RAI No. 422, FSAR Ch. 3, Supplement 21," June 1, 2011.

Ref. 24: E-mail, Dennis Williford (AREVA NP Inc.) to Getachew Tesfaye (NRC), "Response to U.S. EPR Design Certification Application RAI No. 422, FSAR Ch. 3, Supplement 22," June 6, 2011.

Ref. 25: E-mail, Dennis Williford (AREVA NP Inc.) to Getachew Tesfaye (NRC), "Response to U.S. EPR Design Certification Application RAI No. 422, FSAR Ch. 3, Supplement 23," June 24, 2011.

Ref. 26: E-mail, Dennis Williford (AREVA NP Inc.) to Getachew Tesfaye (NRC), "Response to U.S. EPR Design Certification Application RAI No. 422, FSAR Ch. 3, Supplement 24," July 7, 2011.

Ref. 27: E-mail, Dennis Williford (AREVA NP Inc.) to Getachew Tesfaye (NRC), "Response to U.S. EPR Design Certification Application RAI No. 422, FSAR Ch. 3, Supplement 25," July 12, 2011.

AFFIDAVIT

COMMONWEALTH OF VIRGINIA)
) ss.
COUNTY OF CAMPBELL)

1. My name is Sandra M. Sloan. I am Manager, New Plants Regulatory Affairs for AREVA NP Inc. and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by AREVA NP to determine whether certain AREVA NP information is proprietary. I am familiar with the policies established by AREVA NP to ensure the proper application of these criteria.

3. I am familiar with the AREVA NP information contained in "Response to Request for Additional Information No. 422, Revision 0, Supplement 26," and referred to herein as "Document." Information contained in this Document has been classified by AREVA NP as proprietary in accordance with the policies established by AREVA NP for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by AREVA NP and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is

requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information".

6. The following criteria are customarily applied by AREVA NP to determine whether information should be classified as proprietary:

(a) The information reveals details of AREVA NP's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for AREVA NP.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for AREVA NP in product optimization or marketability.

(e) The information is vital to a competitive advantage held by AREVA NP, would be helpful to competitors to AREVA NP, and would likely cause substantial harm to the competitive position of AREVA NP.

The information in the Document is considered proprietary for the reasons set forth in paragraphs 6(b) and 6(d) above.

7. In accordance with AREVA NP's policies governing the protection and control of information, proprietary information contained in this Document has been made available, on a limited basis, to others outside AREVA NP only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. AREVA NP policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

Sandra M. Sloan

SUBSCRIBED before me this 29th

day of July , 2011.

Kathleen A. Bennett

Kathleen A. Bennett
NOTARY PUBLIC, COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES: 8/31/2015
Registration No. 110864

Notary Public
Commonwealth of Virginia
Aug 31, 2015

Response to

Request for Additional Information No. 422, Supplement 26

8/3/2010

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
Application Section: 3.9.2

QUESTIONS for Engineering Mechanics Branch 2 (ESBWR/ABWR Projects) (EMB2)

Question 03.09.02-88:

Follow-up to RAI 245, Question 03.09.02-59

The applicant responded to RAI 03.09.02-59 in the response to RAI 245 by referencing the CVAP. The staff noted that the analysis of other components within the recirculating steam generator (RSG) is briefly discussed in CVAP Section B.5 together with a statement that the analyses show that the components will not experience excessive flow induced vibration. The staff noted that neither the analyses nor the results are provided and therefore the staff could not determine if RSG components are vulnerable to flow induced vibration. Each issue associated with FIV of RSG components that requires resolution of RAI 03.09.02-59 is described below.

Four units currently operating in Europe have similar upper internals to the U.S. EPR RSG design and have been operating for a significant period of time:

1. 79/19T RSG – Doel unit 4, in service since 1996 (13 years)
2. 79/19T RSG – Tihange Unit 3, in service since 1998 (11 years)
3. 73/19TE RSG of CZB1, in service since 1996 (13 years)
4. 73/19TE RSG of CV2, in service since 1999 (10 years).

The staff noted that the U.S. EPR has a significantly higher mass flow of steam through the RSG (17.8 percent higher than Doel and Tihange units, and 8.5 percent higher than the CZB and CV units). The applicant is request to discuss the significance of the higher flow rate through the steam dryers with respect to nominal velocity, surface area and the similarity of the four comparison plants.

Response to Question 03.09.02-88:

The recirculation flow from the U.S. EPR primary steam separators and dryers is approximately **[ ]** percent of the secondary side flow rate through the tube bundle, as reported in Technical Report ANP-10306P, Comprehensive Vibration Assessment Program for U.S. EPR Reactor Internals, Table B-1. Referring to Figure 03.09.02-88-1, the recirculation flow is drained from the steam dryers in the upper internals through drain piping located in the periphery of the steam drum to an elevation just below the entrance to the steam separators. The majority of the recirculation flow from the steam separators is discharged from the first stage, which is located at the entrance to the separators as shown in Figure 03.09.02-88-1.

Given the large flow area created by the boundary of the steam generator (SG) upper shell and the small magnitude of the single-phase recirculation flow, the SG upper internal components identified in Technical Report ANP-10306P, Section B.5—e.g., the main feedwater distribution system, the emergency feedwater header rings— cannot experience significant flow excitation due to random turbulence from the recirculation flow. The location of the drain piping in the periphery of the steam drum, away from the steam separator exhaust, eliminates the possibility for this flow to create flow excitation of the drain piping due to random turbulence or vortex-shedding.

Although the secondary side mass flow rate for the US EPR steam generator (SG) is [ ] percent higher than the SG identified in Technical Report ANP-10306P, Table B-1, the U.S. EPR SG upper internal components are not susceptible to significant flow excitation for the reasons identified.

As described in Technical Report ANP-10306P, Sections B.3 and B.4.2, the main steam, main feedwater, and other piping systems that are attached to the SG will be instrumented with permanent sensors to measure and monitor pipe vibrations during startup testing and throughout the service life of the plant. If unexpected vibrations from acoustic resonance are observed in the attached piping systems during start-up testing, the sources of excitation will be identified and the piping redesigned as needed.

Technical Report ANP-10306P, Section B.2.1 provides a comparison of the design of the U.S. EPR SG upper internals and the plants referenced in RAI 422, Question 03.09.02-88. A comparison of the secondary side mass flow rate and the densities entering the steam drum, as well as other design aspects, is provided in Technical Report ANP-10306P, Table B-1.

The evaluation of the steam dryer design for flow-induced vibration (FIV) is provided in Technical Report ANP-10306P, Section B.4. The steam dryer design of the four operating plants referenced in RAI 422, Question 03.09.02-88 is essentially identical to the U.S. EPR steam dryer design. As shown in Technical Report ANP-10306P, Table B-1, the flow area for the dryer cells is [ ] percent greater for the U.S. EPR steam dryers. The nominal flow velocity through the 79/19T SG dryer cells is [ ] inch/sec and [ ] inch/sec for the 73/19TE SG designs, providing a lower and upper bound for the U.S. EPR SG steam dryer design at [ ] inch/sec. The temperature and density of the steam in the steam drum is also approximately the same for the referenced plants and the U.S. EPR SG. Theoretically, the modal frequencies and the mode shapes for the steam dryers would be the same for the five plants, with the exception of the slight differences due to the rectangular dimensions of the dryer cells and the resulting influence of the hydrodynamic mass of the steam.

The turbulent excitation of the steam flow through the dryer cells and across the chevron-type dryer vanes is similar among the operating and non-operating SG designs. Based on the comparison of the nominal flow velocity and density, as well as the similar frequency aspects of the STAR design, the response of the U.S. EPR STAR dryers is bounded by the CZB1 and CV2 operating plants.

The combined operating experience for the STAR steam dryers is more than 40 years with at least 10 years of operating experience for each of the four operating plants using the STAR dryer design. Refer to the response to RAI 422, Question 03-09-02-89 for the effect of the economizer.

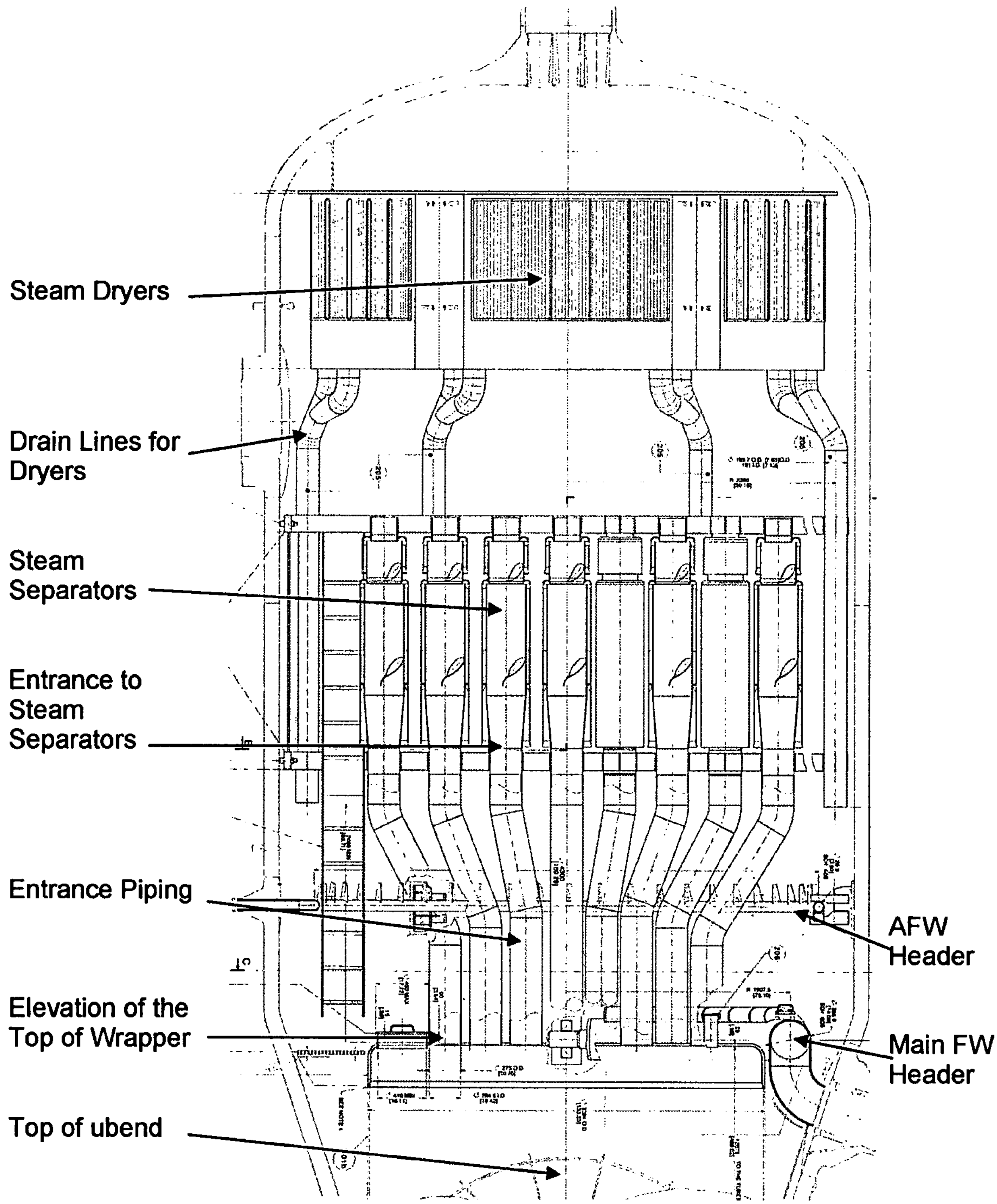
FSAR Impact:

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

Technical Report Impact:

ANP-10306P, "Comprehensive Vibration Assessment Program for U.S. EPR Reactor Internals Technical Report," Revision 0 will not be changed as a result of this question.

Figure 03.09.02-88-1—U.S. EPR SG Upper Internals



Question 03.09.02-131:

This is related to RAI 03.09.02-48d.

The applicant detailed the analysis procedure for fluid-elastic instability in CVAP Section 4.5.1.1.3. The applicant states that the fluid-elastic calculations will be performed for the CRGA but fluid-elastic analysis was not performed for the Normal columns, the LMP and the Instrumentation Guide Tube. The applicant is requested to provide justification for not performing the fluid-elastic instability analysis for the normal columns, the LMP and the instrumentation guide tube.

Response to Question 03.09.02-131:

As stated in the response to RAI 422, Questions 03.09.02-130 and 03.09.02-132, the flow induced vibration (FIV) analysis for the upper column supports will be revised to consider an improved thermal hydraulic prediction or distribution of primary flow in the upper internals. The revised analysis for fluid-elastic instability of the normal column, level measurement probe (LMP) column, instrumentation guide tube, and control rod guide assembly (CRGA) tube incorporate the improved thermal hydraulic predictions in the upper internals. The revised results for the analysis of fluid-elastic instability will be incorporated into Technical Report ANP-10306P, Table 4-20.

The analysis of the upper internals for flow excitations resulting from random turbulence and vortex-shedding has also been revised to incorporate the improved thermal hydraulic predictions. The revised results will be incorporated into Technical Report ANP-10306P, Table 4-19 and Table 4-20.

Additional figures will be added to Technical Report ANP-10306P, Section 4.5.3, to provide the frequency and mode shapes plots and the response power spectral density (PSD) curves for each of the support columns and the instrumentation guide tube.

Technical Report ANP-10306P, Section 4.5 will be revised to address changes to the upper internals analysis.

FSAR Impact:

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

Technical Report Impact:

ANP-10306P, "Comprehensive Vibration Assessment Program for U.S. EPR Reactor Internals Technical Report," Revision 0 will be revised as described in the response and indicated on the enclosed markup.

Question 03.09.02-134:

This is related to RAI 03.09.02-48d.

The applicant is requested to review (and correct) a possible discrepancy in the definitions of the Random Lift Coefficient given in the CVAP text on pg 4-131, where it is listed as having the value of 0.01 for f > 120 Hz, while in CVAP Figure 4-34, it is listed as possessing a value of 0.01 for f = 65 Hz. The staff noted that this correction is especially important since the analysis of the turbulent buffeting extends above 300 Hz for the instrumentation guide tube.

Response to Question 03.09.02-134:

The random lift coefficient defined by Technical Report ANP-10306P, Figure 4-34, for the frequency range of 0 to 65 Hz is correctly transcribed from its source reference. Refer to the response to RAI 422, Question 03.09.02-131 and the changes to the Technical Report ANP-10306P incurred by that question for a more detailed definition of the PSD that is applied to the structural components of the upper internals. In the response to Question 03.09.02-131, note that an editorial change to the definition of the PSD is identified in the Technical Report ANP-10306P, Section 4.5.1.1.4. Specifically, the random lift coefficient for frequencies greater than 120 Hz was changed from 0.01 to 0.00133 to be consistent with the definition of the PSD provided in the Technical Report ANP-10306P, Figure 4-34. The lower magnitude of the analysis results for random turbulence that are provided with Question 03.09.02-131 are solely due to the revised thermal hydraulic loading which is also lower. The analytical evaluation of the column supports for random turbulence considered the corrected definition for the PSD that is identified in the revised Technical Report ANP-10306P, Section 4.5.1.1.4.

FSAR Impact:

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

Technical Report Impact:

ANP-10306P, "Comprehensive Vibration Assessment Program for U.S. EPR Reactor Internals Technical Report," Revision 0 will not be changed as a result of this question.

Question 03.09.02-140:

Follow-up to RAI 245, Question 03.09.02-53

The staff issued RAI 03.09.02-53 as a follow-up to RAI Question 03.09.02-34. 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, RAI 03.09.02-53 is initiated requesting the program with a description of the nondestructive testing.

The applicant responded to RAI 03-09.02-53 in their Response to Request for Additional Information No. 245, Supplement 4. The applicant stated that 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, 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. The applicant also stated in Section 6.1 that non-destructive testing is not planned for the inspection of the RV or the RV internals.

The staff noted that Section 2.3 (3) of Reg. Guide 1.20 (Rev-3) recommends that an applicant provide description of the inspection procedure, including the method of examination (e.g., visual and nondestructive surface examinations), method of documentation, provisions for access to the reactor internals, and specialized equipment to be employed during the inspections to detect and quantify evidence of the effects of vibration.

The applicant is requested to explain why they have excluded the non-destructive testing as recommended by Reg. Guide 1.20, (particularly surface examinations) when it is known that visual inspections will not be sufficient to reveal any cracks that might have initiated due to Flow Induced Vibrations during the HFT lasting a duration to fatigue cycle the RV internals to10^6 cycles.

Response to Question 03.09.02-140:

The U.S. EPR reactor pressure vessel (RPV) and the internals visual inspections will meet the guidelines and requirements of ASME Section III, Paragraph NG-5111, 2004 and the methods defined in ASME Section V, Article 9. The visual inspections “VT-1” and “VT-3” required by ASME Section XI, Subsection IWB-2500, Table IWB-2500-1 for the examination categories of the reactor vessel (RV) and core support structures will be followed to meet the requirements of Regulatory Guide 1.20, Revision 3, inspection program. The acceptance criteria for these nondestructive surface examinations defined in ASME Section XI, Subsection IWB-2500, Table IWB-2500-1 will be used to inspect the surfaces and welds of the components identified in Technical Report ANP-10306P, Table 6-1 through Table 6-5. See Table 03.09.02-140-1 for a summary of the visual inspection plan.

Technical Report ANP-10306P, Section 6.1 will be revised to reflect these changes to the inspection program.

FSAR Impact:

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

Technical Report Impact:

ANP-10306P, "Comprehensive Vibration Assessment Program for U.S. EPR Reactor Internals Technical Report," Revision 0, Section 6.1 will be revised as described in the response and indicated on the enclosed markup.

Table 03.09.02-140-1—U.S. EPR Visual Inspection Plan (per 2004 ASME Section XI, Table IWB-2500-1)

Parts Examined	Examination Category	Examination Method	Acceptance Criteria	Extent
Interior of Reactor Vessel	Category B-N-1	Visual, VT-3	IWB-3520.2	Surface
Interior Attachments to Reactor Vessels	Category B-N-2	Visual, VT-1	IWB-3520.1	Welds
Welded Core Support Structures	Category B-N-2	Visual, VT-1	IWB-3520.1	Welds
Removable Core Support Structures	Category B-N-3	Visual, VT-3	IWB-3520.2	Surface

Notes for Table 03.09.02-140-1:

1. The ASME Code, Section XI, Table IWB-2500-1 prescribes a VT-3 examination method for the B-N-2 examination category of the welded core support structures. In order to provide a more rigorous examination method, above that which is required by the ASME code for these parts, the examination method VT-1 and the corresponding acceptance criteria (IWB-3520.1) will be followed for the pre-service inspection of the RPV internals following hot function testing.

ANP-10306P Markups

- Random turbulence induced vibration.

The full scale theoretical analysis of these FIV mechanisms is performed considering a range of compressive axial loads to evaluate the sensitivity of the column supports to these sources of flow excitation.

The column supports are evaluated for the thermal hydraulic conditions representative of full power steady state operating conditions. The steady state flow conditions are augmented to effectively evaluate the 10 percent RCP overspeed transient condition by conservatively increasing the flow velocities by 10 percent (relative to the full power steady state normal operating condition) while still considering the high cycle fatigue associated with 60 EFPY with a 100 percent capacity factor.

The requirements of RG 1.20 (Reference 1) require that the transient conditions associated with RCP transients be evaluated. For these transients, flow reversal through one to three hot leg nozzles will occur, depending upon the combinations of RCPs that are being operated. The largest flow reversal, in terms of volume or flow velocity through the hot leg nozzle, occurs with three (3) RCP operating at the 75% power level. However, the largest dynamic pressure occurs with three (3) RCP operating at cold shutdown condition (370 psia, 140F). Since it is the dynamic pressure term that is the most appropriate parameter to compare, the RCP transient at cold shutdown was evaluated.~~The misdistribution of flow that occurs in the RV downcomer and lower plenum with various combinations of RCP operation redistributes itself upon entering the RV upper plenum. With the exception of the two RCP operation, the FIV response of the upper internals is bound by the operating configuration with four RCP operation because of the greater volume of flow and turbulence that occurs in the upper plenum. The FIV evaluation considering the thermal hydraulic conditions associated with four RCP operations is performed.~~

03.09.02-131

4.5.1.1 Analysis Methodology and FIV Design Inputs

4.5.1.1.1 Modal Analysis

The natural frequencies of the column supports are determined using the structural analysis program CASS with three dimensional beam elements. The frequencies and eigenvectors of the column supports are determined in the water environment at the full power operating temperatures. The effects of the hydrodynamic mass of the primary fluid on the in-air natural frequencies of the column support are accounted for by calculating an equivalent specific mass using the following formula:

[ ]

For an infinite cylindrical tube, the displaced water corresponds to the water contained by the inner surface of the cylinder and the water displaced by the outer surface of the cylinder. The hydrodynamic mass per unit length of the displaced water is then:

[ ]

4.5.1.1.2 Vortex-Shedding Induced Vibrations

All three column types and the instrumentation guide tubes are evaluated for their susceptibility to experience vortex shedding induced vibration. The vortex shedding frequencies of these structures are determined from the relation:

$$f_s = \frac{S_t V}{D}$$

Where:

V = the free stream flow velocity approaching the cylinder of a diameter "D."

S_t = the dimensionless Strouhal number, which is dependant upon the Reynolds number.

The correlation between the Strouhal number and the Reynolds number is shown in Figure 4-33 for a single cylinder exposed to cross flow conditions (e.g., the instrumentation guide tube). For the other column types, the Strouhal number for the array of column supports is determined from the correlation shown below for a triangular array of cylinders, which is taken from Reference 4, Section 6.6:

$$S = \frac{1}{1.73(P/D-1)}$$

Where:

03.09.02-131

P/D = the pitch to diameter ratio.

This expression for the Strouhal number is based on the approach velocity rather than the pitch velocity for which the thermal hydraulic data from the CFD model is provided. The Strouhal number expressed in terms of pitch velocity is:~~For the three different types of column supports, the P/D ratio is determined for the CRGA array of column supports because these supports dominate the flow characteristics of the upper internals.~~

$$S_p = \frac{1}{1.73(P/D)}$$

For cases of off resonances between the vortex shedding frequencies and the structural frequencies, the response of the column supports and the guide tube(s) to vortex-shedding excitation was determined through relationships for forced harmonic vibrations, considering the forcing functions in both the lift (F_L) and drag (F_D) directions given by:

$$F_L = (\rho V^2/2)(D)(C_L)Sin(2\pi f_L t)$$
$$F_D = (\rho V^2/2)(D)(C_D)Sin(2\pi f_D t)$$

The amplification factor, "A_o" of these dynamic loads is equal to:

$$A_o(f) = \frac{1}{\{[1-(f/f_o)^2]^2 + 4\zeta^2(f/f_o)^2\}^{1/2}}$$

03.09.02-131

For closely spaced arrays of cylinders with the pitch to diameter (P/D) spacing less than about 1.5, the distinct frequency associated with vortex-shedding degenerates into broadband turbulence. For the CRGA array of cylinders, the P/D ratio is equal to 1.26. The span wise variation of the velocity profile along the length of the support columns also reduces the strength of the lock-in. Nonetheless, the support columns and the instrumentation guide tubes were evaluated for lock-in using the design methods ~~The design methods~~ outlined in Reference 9a ~~are followed~~ to verify that a resonance condition between the natural frequencies of the column supports and the vortex shedding frequencies of the supports is avoided and that lock-in conditions will not occur. For a single cylinder (e.g., the instrumentation guide tube), lock-in can be avoided by any one of the four following methods. For an array of cylinders (the three column support type), only the first three bullets are applicable.

The design criteria used to verify that the lock-in condition is suppressed for these structures are:

a. If the reduced velocity for the fundamental vibration mode (n=1) satisfies the condition: $\frac{V}{f_n D} < 1$, then both the lift and drag directions lock-in are avoided.

b. If the reduced damping (for a given vibration mode) is larger than 64: $C_n > 64$, then lock-in will be suppressed for that vibration mode.

c. For a given mode of vibration, the two following conditions are verified: $\left\{\begin{matrix} V/f_n D < 3.3 \\ C_n > 1.2 \end{matrix}\right\}$ then the lift direction lock-in is avoided and drag direction lock-in is suppressed.

d. For the structural frequencies, verify: $f_n < 0.7 f_s$ or $f_n > 1.3 f_s$, where f_n is the natural structural frequency of the n^{th} mode and f_s is the frequency of the periodic vortex shedding in either the lift or drag directions.

The reduced damping of the n^{th} mode is equal to:

$$C_n = \frac{4\pi \xi_n M_n}{\rho D^2 \cdot \int_{L_e} \phi_n^2(x)dx}$$

Where:

ξ_n = the damping of the n^{th} mode.

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:

$$FSM = V_c / V_p$$

If the FSM is less than 1.0, the array of column supports is predicted to be unstable, while an FSM that exceeds 1.0 indicates that the column supports will be stable. The value of FSM above 1.0 represents a margin of safety in the usual engineering sense. The acceptance criterion for fluid-elastic instability for the U.S. EPR is an FSM greater than 1.3. The fluid-elastic instability ratio (V_P / V_c) is simply the inverse of the AREVA NP definition for FSM.

4.5.1.1.4 Random Turbulence Induced Vibrations

The analysis for random turbulence-induced random vibration, or turbulent buffeting, resulting from cross flow conditions is based on finite element implementation of the acceptance integral method of References 7 and 8 using the computer program PCRANDOM. The methodology implemented by ~~"~~PCRandom~~"~~ to determine the RMS response of the columns supports is discussed in Section 4.3.2.1. 03.09.02-131

The column supports are evaluated as a one dimensional beam based on the conservative assumption that the random pressure is fully coherent across the diameter of the columns. The empirical relationship for the turbulent response of a one dimensional structure, or in this case, the column supports, is defined below and is consistent with the guidelines established in Reference 9a, Paragraph N-1340.

$$< y_n^2 > = \sum_n \frac{L G_F(f_n)\Phi_n^2(x)}{64\pi^3 m_n^2 f_n^3 \varsigma_n} J_{nn} + cross\ terms$$

Where:

$<y_n^2>$ = the mean square vibratory amplitude.

L = the column length.

n = a modal subscript.

$G_F(f_n)$ = the single sided random force PSD.

$\Phi_n(x)$ = the mode shape function along the length of the column.

$m_n = \int_l \Phi_n^2(x)\, m(x)dx$ or the generalized mass.

m(x) = the linear mass (physical + hydrodynamic) along the length of the beam.

ϕ_n = the mode shape function of the n^{th} mode.

L_e = the cylinder length subject to lock-in cross flow.

M_n = the generalized mass of the n^{th} mode or $M_n = \int_0^L m_t(x)\phi_n^2(x)dx$

D = the cylinder diameter.

ρ = the mass density of the primary fluid.

The damping inherent to the column supports is attributed to the viscous damping created from the fluid (i.e., fluid damping due to fluid drag), the structural damping of the material (or hysteresis), and the damping associated with the non-linear interaction between the bolted connections of the columns with the UCP and USP. The damping for the column supports is determined from the following relationship, which is applicable to power plant piping. This correlation was a result of an extensive compilation of damping data of pipe sizes ranging from 1 to 18 inches. A regression analysis was performed on the data provided the damping correlation shown below, which is taken from Reference 2, Equation 8-43.

$$\zeta = 0.0053 + 0.0024D + 0.0166R + 0.009F - 0.019L$$

Where:

D = the diameter of the column support (in inches).

R = the response level, equal to zero (0) if there is no yielding and one (1) if the amplitude is sufficient to cause yielding.

F = 1 for the first mode (N = 1) and 0 for the higher modes.

L = 0 if the piping is relatively uniform and equal to 1 if relatively massive valves or other attached equipment is attached to the piping.

This damping relationship is used to determine the modal damping ratio for the normal, LMP and CRGA column supports as summarized in Table 4-17. An equivalent viscous damping ratio of **[ ]** percent that is associated with hysteresis is applied to the instrumentation guide tube since this component is welded to the supports.

4.5.1.1.3 Fluid-elastic Instability

Reference 9a, paragraph N-1331.2 provides a criterion in which fluid elastic instability can be avoided for a tube bundle. This criterion is applied to the upper internals ~~array of CRGA column supports~~ to predict fluid-elastic instability for the array of ~~the~~ CRGA columns, normal and LMP column supports, and the instrumentation guide tube, and is based on the prediction of a critical velocity that must not be reached or exceeded. Connors' equation predicts that a tube or

03.09.02-131

f_n = the modal frequency in units of Hz.

ζ_n = the modal damping ratio.

J_{nn} = the joint acceptance integrals in the axial direction of the column.

Because the modal frequencies of the columns are separated, there is no contribution from the cross terms or from the dynamic response due to closely spaced modes of the columns.

The single phase PSD function as proposed by Pettigrew and Gorman, recommended by Reference 9a, Figure N-1343-1, and reproduced in Figure 4-34 is applied to the column support types. Typically, this PSD is considered appropriate for tubes in heat exchangers with diameters in the range of ~ ½ and ¾ inches with pitch-to-diameter ratios of approximately 1.4. The minimum pitch to diameter ratio of the CRGA column supports is ([ ]), and this array of cylinders has equivalency in this aspect.

The upstream random lift coefficient from Figure 4-34 is conservatively applied to the column supports and the instrumentation guide tubes. The pressure PSD is determined by multiplying the dynamic pressure by a frequency dependent excitation coefficient from Figure 4-34 as follows:

$G_p(f)$ (pressure PSD) = $[C_R(f)\, q(x)]^2$

q (dynamic head) = $\frac{1}{2}(\rho V^2)$

C_R (random lift coefficient) $= 10^{(af + b)}$

03.09.02-131

$= 0.025$ for $f < 40$ Hz

$= 0.0\underline{0133}\cancel{10}$ for $f \geq 120$ Hz

Where:

a = -15.92E-03.

b = -965.4E-03.

<u>To provide a PSD with a range of frequency that bounds the frequencies of the column supports, the random lift coefficient (C_R) at the 120 Hz frequency as defined above is extended beyond the original source of the PSD that is shown in Figure 4-34. The value of the random lift coefficient (C_R) at the 120 Hz frequency is established using the empirical definition of this PSD provided above. It is the PSD shown in Figure 4-50 which includes the 0 to 120 Hz range of frequency that is actually used for the analysis of the normal column supports and the instrumentation guide tube.</u>

4.5.2 FIV Acceptance Criteria for the Column Supports

4.5.2.1 Fluid-elastic Instability

The FSM is the ratio of the critical velocity where an array of cylinders is predicted to become unstable to the equivalent mode shape weighted pitch velocity. An FSM greater than 1.0 implies that the array of cylinders is stable, while an FSM less than 1.0 implies that the array of cylinders will become unstable. For the U.S. EPR, a conservative acceptance criterion for fluid-elastic instability of FSM greater than 1.3 is used, which represents a margin of safety 30 percent.

4.5.2.2 Random Turbulence-Induced Vibration

The turbulence-induced vibration analysis is based on Powell's joint acceptance technique (Reference 8), which computes only the RMS vibration amplitude and stresses. The displacements are computed in units of (inch, rms), the moments in units of (in-lbs, rms) and reaction loads at support locations in units of (lbs, rms).

<u>Acceptance Criteria for Displacements</u>

Because the computed displacement response to turbulence is based on a probability of excursions, the displacements (determined in units of inch, rms) are multiplied by a factor of five or five sigma, which represents an approximately 100 percent probability that the computed displacement will not be exceeded. The gap clearance between the adjacent CRGA locations is [ ] One-half of this value, [ ] inches, is used as the acceptance criteria to prevent impacts between adjacent CRGAs. Therefore, the allowable displacement in units of inch, rms is [ ] inch/5sigma or [ ] inch, rms for turbulence-induced vibration. Because the transverse displacement of the CRGA column supports will exceed the gap clearance between its ID and the CRGAs, experimental fatigue characterization test of the CRGAs is performed. The results of the test are reported in Section 4.6.2.

The allowable displacement limit of [ ] inch, rms is conservatively applied to the other column supports (normal and LMP) because the clearance between these components and the CRGA column supports is greater, which is conservative.

03.09.02-131

<u>Acceptance Criteria for High Cycle Fatigue</u>

The criterion established in Section 4.2.6.2 using fatigue curve "~~A~~<u>C</u>" is applied to the columns of the upper internals. The allowable stress for fatigue curve "~~A~~<u>C</u>" at 10^{13} cycles is [ ] psi, rms.

4.5.2.3 Vortex-Shedding Induced Vibrations and RCP Acoustic Pressure Fluctuations

Acceptance Criteria for Displacements

The acceptance criterion for the off resonant response of the column supports assures that the mid-span displacements of the column supports are small enough to avoid impact with adjacent column supports. Because the response to vortex-shedding and the RCP acoustic pressure fluctuations is harmonic, the allowable displacement and the allowable stress are in units of 0-peak. As computed for turbulence, the allowable displacement limit for all CRGA column supports is [ ] inch (0-peak).

03.09.02-131

Acceptance Criteria for High Cycle Fatigue

The ASME fatigue curve "~~A~~C" shown in Figure 4-21 is applied to ~~the off lock-in~~ the response of the structure. The allowable high cycle fatigue stress of [ ] (0-peak) at 10^{11} cycles for fatigue curve "~~A~~C."

4.5.3 Response of the Column Supports

The primary fluid velocity and density throughout the RV upper internals is determined with a three dimensional CFD model ~~one-dimensional thermal hydraulic model~~ for the full power normal operating condition. The U.S. EPR upper plenum thermal hydraulic model is qualified based on benchmarking with flow tests performed with the ROMEO mock-up which simulates the upper internals at a scale of 1:5.2. ~~The magnitude of velocity and density that is evaluated for the column supports is representative of a location near the hot leg outlet nozzles (V = [37.6] ft/sec and ρ = [40.8] lbm/ft³). This~~The worst located column support is evaluated. As noted in Section 4.5.1, the cross flow velocity distribution along the length of the column supports is increased by 10 percent to account for the RCP over-speed transient conditions. ~~The distribution of velocity at the hot leg nozzle elevation is projected upon the corresponding flow area of the column supports to provide a velocity profile for the analysis.~~

~~Since the time at which the full scale analysis of the column supports were originally evaluated, the velocity profile along the length of the column supports has been refined using three dimensional thermal hydraulic modeling and analysis of the flow distribution in the upper plenum. AREVA plans to revise the FIV analysis of these column supports to consider the refined velocity distribution and to remove other conservatisms in the current evaluation. The revised analysis is expected to improve the accuracy of the results to provide better agreement between the pre-operational evaluation and the hot functional test results. A preliminary assessment for the column supports, considering the revised thermal hydraulic conditions, shows a lower response to flow excitations resulting from turbulence for the column supports.~~

The FIV analysis of the column supports and the instrumentation guide tube for the limiting RCP transient condition is performed for fluid-elastic instability. The method by which the full power

normal operating conditions are evaluated, considering the 10% RCP transient and a capacity factor of 100% for 60 EFPY, provides a conservative estimate of the total fatigue usage for the high cycle effect created by random turbulence and vortex-shedding such that detailed analysis of this short transient conditions is not necessary.

The full scale theoretical results, considering the sources of excitation identified in Section 4.5.1, are summarized in ~~the following tables~~Table 4-18 through Table 4-20 and Figure 4-51 through Figure 4-58.

03.09.02-131

Table 4-19—Evaluation for Vortex-Shedding Lock-in for the Column Supports

[ ]

~~ASME Design Guidelines to avoid Vortex-Shedding lock-in~~	~~———~~	~~$C_n > 64$~~	~~———~~	~~$f_n < 0.7f_s$ or $f_n > 1.3f_s$~~	~~Evaluation~~
~~Normal Column support~~	~~For modes 4 & 5~~	-	~~For modes 1, 2, 3~~	~~N/A~~	~~Lock-in Avoided~~
~~LMP Column support~~	~~For modes 4 & 5~~	-	~~For modes 1, 2, 3~~	~~N/A~~	~~Lock-in Avoided~~
~~CRGA Column support~~	~~For modes 1 to 5~~	-	-	~~N/A~~	~~Lock-in Avoided~~
~~Instrumentation Guide Tube~~	~~For modes 1 & 2~~	-	-	~~For modes 1 & 2~~	~~Lock-in Avoided~~

03.09.02-131

Note(s) for Table 4-19:

1. The vortex-shedding frequencies of the column support during the limiting RCP transient with three RCPs operating at cold shutdown conditions are bounded by the vortex-shedding frequencies for the 100 percent power normal operating conditions.
2. See Section 4.5.1.1.2 for details regarding the acceptance criteria for the lock-in condition as provided by the ASME Section III, Appendix N, Paragraph N-1324.1.

03.09.02-131

Table 4-20—Summary of FIV Results for the RV Upper Internal Column Supports (100% Power, Steady State, Normal Operating Conditions)

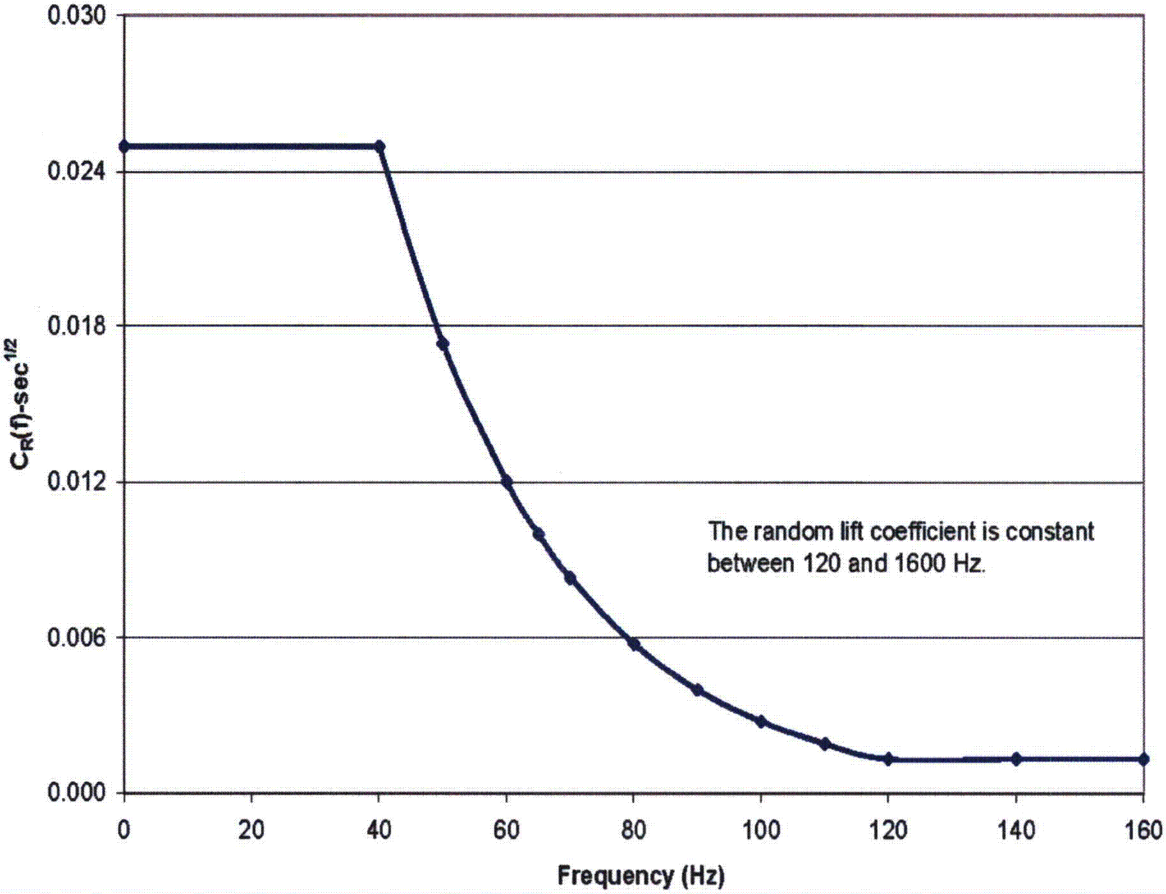
[ ]

Notes for Table 4-20:

1. The stresses reported in this table are based on a conservative FSRF of 3.0 which bounds the computed values for the structural discontinuities of the lower and upper flanges. The full penetration welds used to join the column support parts do not require an FSRF per the ASME Section III requirements; however, a FSRF was conservatively applied.

03.09.02-131

Figure 4-50—Random Lift Coefficient for U.S. EPR Upper Internals



03.09.02-131

Figure 4-51—Mode Shapes for the Normal Support Column (axial load = -4500 lbs

03.09.02-131

Figure 4-52—Mode Shapes for the LMP Support Column (axial load = -3600 lbs)

03.09.02-131

Figure 4-53—Mode Shapes for the CRGA Support Column (axial load = -7500 lbs)

03.09.02-131

Figure 4-54—Mode Shapes for the Instrumentation Guide Tube

[ ]

03.09.02-131

Figure 4-55—Displacement Response PSD for Normal Support Column

[ ]

03.09.02-131

Figure 4-56—Displacement Response PSD for LMP Support Column

03.09.02-131

Figure 4-57—Displacement Response PSD for CRGA Support Column

03.09.02-131

Figure 4-58—Displacement Response PSD for Instrumentation Guide Tube

[ ]

03.09.02-131

6.0 INSPECTION PROGRAM

6.1 Principles of the Inspection Program

The inspection program for the RV internals involves visual inspections before and after the pre-operational tests. The internals are removed from the RV and placed on a storage stand in the reactor cavity for inspection. The accessible areas of the RV internals that are visually examined are:

- The fastening devices.
- The bearings surfaces.
- The interfaces between the RV internal components that are most likely to reveal relative motions and wear.

The inside of the RV is also visually examined for abnormalities.

The RV internals will be visually examined with 5 to 10X magnification. Non-destructive surface examinations will be used to inspect the welds of the RV internals. ~~Non-destructive testing is not planned for the inspection of the RV or the RV internals.~~

The integrity of the RV internals is considered adequate and passes the inspection program phase of the comprehensive vibration assessment program if no indication of abnormally large vibration amplitudes or excessive wear is detected.

03.09.02-140

6.2 Inspection Plan

Visual inspections of the U.S. EPR RPV and its internals will adhere to the guidelines and requirements provided by the 2004 edition of the ASME Section III, Paragraph NG-5111 and the methods defined in the ASME Section V, Article 9. The visual inspections "VT-1" and "VT-3" required by ASME Section XI, Subsection IWB-2500, Table IWB-2500-1 for the examination categories of the reactor vessel and the core support structures will be followed to fulfill the requirements of the inspection program of RG 1.20. The acceptance criteria for these nondestructive surface examinations are provided in Table 6-6 and will be used to inspect the surfaces and welds of the components identified in Tables 6-1 through 6-5.

Considering that the RV lower and upper internals are stored on the same stand, the inspections shall be performed in the following five steps:

1. Upper internals on the storage stand.
   Inspection of the whole accessible areas of the upper internals. Table 6-1 details the necessary inspections.
2. Lower internals in the RV.
   Inspection of the inside of the lower internals. Table 6-2 details the necessary inspections.
3. Lower and upper internals on the storage stand.
   Inspection of the outside of the lower internals. Table 6-3 details the necessary inspections.

Table 6-4—Visual Inspection of the RV Inside

Component	Sub-component	Inspection
RV 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
	Stellited surfaces of the inserts	Surface aspect

Table 6-5—Visual Inspection of the RV Head

Component	Sub-component	Inspection
RV head flange	Contact surface with the upper internal flange	Surface aspect
Adaptors	Thermal sleeves	Amplitude of vertical displacement

Table 6-6—U.S. EPR RPV and Internals Visual Inspection Plan

Parts Examined	Examination Category	Examination Method	Acceptance Criteria	Extent
Interior of Reactor Vessel	Category B-N-1	Visual, VT-3	IWB-3520.2	Surface
Interior Attachments to Reactor Vessels	Category B-N-2	Visual, VT-1	IWB-3520.1	Welds
Welded Core Support Structures	Category B-N-2	Visual, VT-1	IWB-3520.1	Welds
Removable Core Support Structures	Category B-N-3	Visual, VT-3	IWB-3520.2	Surface

Note(s) for Table 6-6:

1. The ASME Code, Section XI, Table IWB-2500-1 prescribes a VT-3 examination method for the B-N-2 examination category of the welded core support structures. To provide a more rigorous examination method, beyond that required by the ASME code for these parts, the examination method VT-1 and the corresponding acceptance criteria (IWB-3520.1) will be followed for the pre-service inspection of the RPV internals following hot function testing.

03.09.02-140