AREVA

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NRC:10:114

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

Response to U.S. EPR Design Certification Application RAI No. 407, Supplement 3

Ref. 1: E-mail, Getachew Tesfaye (NRC) to Martin C. Bryan (AREVA NP Inc.), "U.S. EPR Design Certification Application RAI No. 407 (4654), FSAR Ch. 3," June 7, 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. 407, FSAR Ch. 3," June 30, 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. 407, Supplement 1, FSAR Ch. 3," October 20, 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. 407, Supplement 2, FSAR Ch. 3," November 22, 2010.

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. 407. Reference 3 provided a revised schedule for technically correct and complete responses to allow additional time to interact with NRC. Reference 4 provided a revised schedule for technically correct and complete responses to allow additional time to address NRC comments.

The enclosed response provides technically correct and complete responses to 8 of the 13 questions. 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.

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

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NRO

Question #	Start Page	End Page
RAI 407 — 03.09.02-70	2	3
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The schedule for the technically correct and complete responses to the remaining 5 questions is unchanged and is provided below.

Question #	Response Date
RAI 407 — 03.09.02-69	January 19, 2011
RAI 407 — 03.09.02-71	January 19, 2011
RAI 407 — 03.09.02-73	January 19, 2011
RAI 407 — 03.09.02-76	January 19, 2011
RAI 407 — 03.09.02-78	January 19, 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

AFFIDAVIT

COMMONWEALTH OF VIRGINIA)
) ss.
COUNTY OF CAMPBELL)

1. My name is Sandra M. Sloan. I am Manager, Regulatory Affairs for New Plants, 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 letter NRC:10:114, "Response to U.S. EPR Design Certification Application RAI No. 407, Supplement 3," 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(c) 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 21st

day of December, 2010.

Kathleen A. Bennett

Kathleen A. Bennett
NOTARY PUBLIC, COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES: 8/31/2011
Reg. #110864

KATHLEEN ANN BENNETT
Notary Public
Commonwealth of Virginia
110864
My Commission Expires Aug 31, 2011

Response to

Request for Additional Information No. 407(4654), Revision 0, Supplement 3

6/07/2010

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 03.09.02-70:

ANP-10306P, Figures 2-1 through 2-2.5

The staff noted that the applicant did not provide weld information, related to type of weld i.e full penetration, double groove etc, and its code acceptability, that should be included in Figures 2-1 through 2-5. The weld information required is described below:

a. Weld between the Core Barrel Flange and Core Barrel Upper Skirt

b. Weld between the Core Barrel Upper Skirt and Core Barrel Lower Skirt

c. Weld between the Core Barrel Lower Skirt and the Lower Support Plate tongue (unknown thickness)

d. Any longitudinal welds in the Core Barrel Flange, Core Barrel Upper Skirt and Core Barrel Lower Skirt

e. Weld between the Upper Support Flange Skirt and the Upper Support Flange

The applicant is requested to explain how each weld configuration identified above in letters (a) through (e) is in accordance with Article NG-3352 of the ASME code and identify the design factors, both 'n' for static loading and 'f' for fatigue loading, per Table NG-3352-1 used in stress calculations for each weld.

Response to Question 03.09.02-70:

[ ]

The full penetration, [ ] welds on the CB and the upper internal hat will be inspected with either RT or UT and PT or MT examination methods to provide a weld quality factor [ ] as outlined in Table NG-3352-1. The weld quality factor [ ] applied to the cyclic stress is applicable to the welds on the CB and the upper internal hat.

The analytical evaluation of the reactor pressure vessel (RPV) lower internals applies a fatigue strength reduction factor (FSRF) of [ ] to the peak stress in the CB, which is entered into the root mean square (RMS) fatigue curve to determine the allowable number of cycles. As explained in Technical Report ANP-10306P, Section 4.2.6.2, this RMS fatigue curve is derived from the ASME Section III fatigue curves provided in Table I-9.0 for austenitic materials. The FSRF of [ ] accounts for the stress amplification effects created by the structural discontinuities of the hot leg nozzles and the CB flange, and conservatively accounts for the weld quality factor for dynamic loads [ ]. See the Response to Questions 03.09.02-79, 03.09.02-80, and 03.09.02-81 for additional information and the FSRF.

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 03.09.02-72:

ANP-10306P, Para 2.1.1 (page 2-2)

The staff noted that Figures 2-1 through 2-5 appear to indicate that bolting is used at the following locations:

- Connection between the Upper Control Rod Guide Assemblies and the Upper Support Plates
- Connection between the Lower Control Rod Guide Columns and the Upper Support Plates
- Connection between the Flow Distribution Device and the Lower Support plate

The staff did not find a description of how the bolts are included in the model or the analysis that was performed to demonstrate bolting integrity is maintained for the required design life. The applicant is requested to provide details of the bolting used in the reactor internals and describe how bolts were addressed in the modeling, including the evaluation of fatigue loading and effects of loss of pre-compression on bolted joint integrity.

Response to Question 03.09.02-72:

The preload in the bolting connections sufficiently prevents separation of these members during the flow-induced vibration (FIV) loadings to these components. The restraint created by these bolted connections was modeled with the appropriate translational and rotational boundary conditions. The comparison of the reaction loads at these restrained locations, which are created from the dynamic loads resulting from the flow excitation of the RPV internals, to the magnitude of preload in the bolted connections, confirms the two mating surfaces will not separate. Additional mechanical excitation and cyclic loading of the RPVI components and/or the bolted connection resulting from separation between the clamped members is not plausible. These bolted connections will not experience the cyclic loadings from FIV load sources that require analysis for high cycle fatigue.

The detailed stress analysis that will address the integrity of the bolting connections to ASME Code Section III requirements considers appropriate load sources and categories, including the effects of loss of pre-compression, and is described in U.S. EPR FSAR Tier 1, Table 2.2.1-5, Item 3.16.

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 03.09.02-74:

ANP-10306P, Para 4.2.2.3 (page 4-16)

The applicant stated that static tests were performed on the HYDRAVIB mock-up to verify that the stiffness of the lower internal assembly and its restraint at the CB flange is modeled accurately. The staff did not find a description of how the stiffness was measured during static testing. Therefore, the applicant is requested to explain how stiffness was measured, and to discuss the comparison of the measured stiffness value with the stiffness value obtained in the finite element model.

Response to Question 03.09.02-74:

The restraint at the core barrel (CB) flange and the bending stiffness of the lower internal assembly were measured by laterally displacing the assembly at the lower support plate (LSP) elevation and recording the applied load and resulting displacement at this elevation. The load and displacement measurements were performed for a range of LSP displacements between 0 and 0.002 inches in the two transverse directions to yield a linear relationship between the stiffness and the displacement and confirmed the absence of any spurious gaps within the mockup. A mean stiffness value equal to [ ] or [ ] was obtained.

From the numerical model of the mockup, the stiffness values were computed along each axis considering the same loading scheme as follows:

K_{NS}= [ ] (for the north-south direction).

K_{EW}= [ ] (for the east-west direction), for a mean stiffness value of [ ].

The difference in stiffness between the two translational directions is due to the presence of the hot leg nozzles.

The mean stiffness values obtained through finite element model (FEM) calculation correlate closely with those obtained through the experimental observations.

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 03.09.02-75:

ANP-10306P, Para 4.2.2.3 (page 4-17)

The applicant stated that shell type modes do not produce significant motion of the LSP or the fuel bundle. As shown in Table 4-2, the shell modes are not identified. The staff could not confirm that the FEA model accurately reflects the shell modes. Therefore, the applicant is requested to describe how the shell modes of the HYDRAVIB scale model assembly compare to those determined by the finite element analysis.

Response to Question 03.09.02-75:

A response to this question will be provided with the Response to RAI 422, Question 03.09.02-116.

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 03.09.02-77:

ANP-10306P

The staff noted that Reg. Guide 1.20 Section 2.1, recommends that the conservatism of simulation of boundary conditions shall be considered in the modeling. Industry experience indicates that there is a strong influence from boundary conditions on the vibrations of the reactor core barrel. The staff did not find a description of the conservatisms used for boundary conditions. Therefore, the applicant is requested to provide details on the boundary condition conservatisms that are applied to the finite element model both for the scale model as well the full FEM model. It is recommended that the applicant provide an ANSYS picture of the finite element model that illustrates the boundary conditions.

Response to Question 03.09.02-77:

The U.S. EPR RPV internals are designed so that the core barrel (CB) flange is clamped by the mating surfaces between the reactor vessel (RV) and the RV head. The cantilever motion of the CB pendulum assembly is limited by the **[ ]** located at the lower support plate (LSP) elevation as described in Technical Report ANP-10306P, Section 2.1.3. This restraint was conservatively not included in the analytical evaluation.

The boundary conditions imposed on the finite element model (FEM) at the CB flange juncture were modeled through a series of one dimensional translational spring elements. The stiffness of the elements was derived through characterization tests on the hold-down ring. To verify the accurate behavior of the model in the numerical model, the stiffness values were confirmed by the experimental simulations described in the Response to Question 03.09.02-74. No other structural boundary conditions were imposed on the model of the reactor pressure vessel (RPV) lower internals.

AREVA NP used a conservative analytical method to model the restraint of the CB flange design. Applying an ideal "clamped boundary condition" at this location represents an infinite stiffness of the spring elements, which increases the natural frequencies of the CB lower assembly and results in a reduction in the response of the CB to turbulence. This conservatism in the boundary conditions was applied to both the scale and full scale analytical models. Figure 03.09.02-73-3 shows the FEM and illustrates the boundary conditions applied to the CB flange to restrain its motion.

Technical Report ANP-10306P, Section 4.2.2.3 will be revised to include a discussion of conservatism in the boundary conditions used in the models.

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 03.09.02-79:

ANP-10306P, Para 4.2.7.2 (page 4-94)

The applicant concluded that high cycle fatigue failure resulting from the random turbulence in the RV downcomer is not likely to occur based on a calculated maximum stress of 1.921 MPa rms. The staff noted that the applicable rules for analyzing fatigue and structural integrity for reactor vessel internals are given in ASME B&PV Code, Section III, Subsection NG, "Core Support Structures – Rules for Construction of Nuclear Facility Components". The stress criteria that must be satisfied for demonstrating compliance with the ASME Code are as shown in TABLE NG 3221-1 and Appendix-D. The applicant is request to demonstrate that the maximum calculated stress satisfies ASME Section III, Subsection NG.

Response to Question 03.09.02-79:

As described in Technical Report ANP-10306P, Section 4.2.7.2, the maximum alternating stress of [ ], adjusted by a fatigue strength reduction factor (FSRF) of [ ] and a scaling factor of [ ] to account for differences in the power level, results in a peak stress of approximately [ ]. For categories of stress that are listed in ASME Section III, Figure NG-3221-1 (e.g., P_m, $P_m + P_b$, $P_m + P_b + Q$ and $P_m + P_b + Q + F$), the maximum peak stress range ($P_m + P_b + Q + F$ = [ ]) from the cyclic loading meets the requirements of ASME Section III, NG-3222.4 for an elastic fatigue evaluation. Because the elastic primary plus secondary stress range for this alternating stress ([ ]) does not exceed the $3S_m$ stress limit, the K_e factor defined in NG-3228.3 is equal to 1.0.

For the other categories of stress identified in ASME Section III, Figure NG-3221-1, the maximum stresses resulting from the flow-induced vibration (FIV) loading on the core barrel (CB) is negligible ([ ]) and these stresses do not need to be compared to the allowable stresses.

A FSRF of [ ] was applied to the alternating stress to account for the structural discontinuity associated with the CB shell and flange where the [ ] stress was computed. A weld quality [ ] for dynamic loading, as prescribed by ASME Section III, Table NG-3352-1 for the [ ] weld joint, was applied to the full penetration weld that is located in this vicinity of the CB shell flange. The weld quality factors for the other girth welds located on the CB assembly (between the upper and lower CB shells and the lower shell to lower support plate (LSP)) are either [ ] welded joints. The peak stress at these locations is bounded by the values identified in this response. There are no welds between the hot leg nozzles and the CB shell. The FSRF for the structural discontinuity of the hot leg nozzle and CB shell and the stress at this location is bounded by the [ ] value at the CB flange. See the Response to Question 03.09.02-81 for information regarding the application of the FSRF to the nominal stress as opposed to the application of a stress concentration factor (SCF) per ASME Section III, NG-3222.4(e)(2).

The correction factor on the alternating stress required by ASME Section III, NG-3222.4(e)(4) to account for differences in the elastic modulus of the RPV internals components at temperature ([ ]) and the ASME fatigue curve at 70°F (28.3 x10^6 psi) was not applied in the fatigue evaluation. The justification for this is provided below.

The primary plus secondary stress range that would be created from the thermal and pressure transients of the Level A and B loading conditions were not included with the alternating range of stress created by FIV ([ ]). These high cycles of stress associated with FIV were ranged with itself, considering a 100 percent capacity factor and 60 effective full power years (EFPYs) to provide a conservative number of design cycles for the stress. It is the primary plus secondary mean stress created with the Level A and B loading conditions that effect the appropriate ASME fatigue curve (A, B, or C).

Referring to the flow chart provided in ASME Section III, Appendices, Figure I-9.2.3, for an elastic analysis at a stress location in three wall thickness with the (P_L + P_b +Q) stress range > 27.2 ksi, the fatigue curve "C" is chosen for comparison. Following the methods discussed in Technical Report ANP-10306P, Section 4.2.7.2 and using fatigue curve "C," as opposed to fatigue curve "A" (refer to the Response to Question 03.09.02-78 for an explanation of fatigue curve "C"), the number of allowable cycles for an alternating stress of [ ] is greater than [ ] because this stress is below the endurance limit at [ ] cycles. The usage factor due to random turbulence in the downcomer at the most limiting location is much less than [ ].

Applying the correction factor (1.11) for the modulus of elasticity to the alternating stress range would result in the same conclusion that the usage factor associated with the high cycle FIV loadings for this component is insignificant and does not need to be evaluated cumulatively with the contribution of fatigue usages incurred with other Level A & B pressure and thermal transients.

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 03.09.02-80:

ANP-10306P, Para 4.2.7.2 (page 4-94)

The staff noted that the Core Barrel has the following different thicknesses:

a. Core Barrel Flange at joint

b. Core Barrel Upper Skirt

c. Core Barrel Lower Skirt

d. Lower Support Plate tongue(unknown thickness)

The staff could not determine from the description of the finite element model in Sections 4.2.2.1 or 4.2.5 or in the ANSYS pictures provided in Figures 4.6 through 4.9 if different thicknesses have been included in the FEM. The applicant is requested to explain how the determination of the maximum stress included the effects of different skirt thicknesses, manufacturing tolerances, stress concentrations from geometric discontinuities, and the effects of welds on the maximum calculated stress.

Response to Question 03.09.02-80:

The finite element model (FEM) of the lower and upper shells of the core barrel (CB) uses the nominal (design) dimensions of each shell. The lower support plate (LSP), which is welded to the bottom of the lower CB, was modeled as a solid circular plate with the density and stiffness adjusted to obtain the same bending modal frequency that would be obtained with a more detailed local model of the LSP. The nominal (design) dimensions of the CB flange were modeled. As described in the Response to Question 03.09.02-74, the hold-down spring was modeled as a series of spring elements with properties derived from numerical and experimental simulations to verify accurate behavior of this aspect of the model.

The components of the CB were modeled with the geometric irregularities of the hot leg nozzles and the CB flanges, with the exception of the local region on the CB flange where the alignment pins are located. This detail was not included in the numerical model. The application of a fatigue strength reduction factor (FSRF) of [ ] conservatively accounted for the stress amplification effects of the local structural discontinuities of the hot leg nozzles and the CB flange juncture. See the Response to Question 03.09.02-81 for additional details concerning the value of the FSRF.

The CB has three girth welds of [ ] (full penetration), which will be inspected using either RT or UT and PT or MT methods, and which conform to the guidelines established in ASME Section III, NG-3351. For this category of welded joints, a dynamic factor [ ] is prescribed by ASME Section III, Table NG-3352-1. Therefore, the nominal stress that would be computed at these weld locations is not amplified by this stress riser and the design life of these welds is not limited by higher dynamic load factors.

Based on the margin for the most limiting stress in the CB ([ ] versus an allowable stress of [ ]), the manufacturing tolerances for the CB would not impact the overall results or conclusions of the current analytical evaluation. See the Response to Question 03.09.02-71 for additional information regarding the influence of manufacturing tolerances on the analytical results. Because the verification of the flow-induced vibration (FIV) results and the integrity of the U.S. EPR RPVI is confirmed with hot functional testing, and RG 1.20 requires that the analytical model be revised as needed to agree with this testing, the current assessment of the maximum stress in the CB is within the acceptance criteria that is established between the analytical and testing simulations as described in Technical Report ANP-10306P, Section 5.5.

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 03.09.02-81:

ANP 10306P, Para 4.2.7.2 (page 4-94)

The applicant applied a fatigue strength reduction factor (FSRF) of 3.0 for the cylinder to flange junction in the calculation of FIV stress. The staff did not find a justification or definition of the FSRF term. The applicant is requested to explain the basis for the fatigue strength reduction factor and why it is applicable for determining FIV stress levels.

Response to Question 03.09.02-81:

The fatigue strength reduction factor (FSRF) is a stress intensification factor that accounts for the effects of local structural discontinuity on the fatigue strength. As described in Technical Report ANP-10306P, Section 4.2.7.2, the FSRF used for the structural discontinuity of the core barrel (CB) to flange juncture is applied to the alternating stress intensity to reduce the fatigue strength. This term differs from a stress concentration factor (SCF), which is applied to the normal component of stress.

Because the majority of the stress intensity ([ ]) is a longitudinal stress created from the pendulum mode of the CB, the FSRF is essentially an SCF. The peak stress of [ ], which was computed using an FSRF of [ ], is a conservative stress and an appropriate method to determine the allowable number of cycles from the ASME fatigue curve.

The SCF computed for this structural discontinuity based upon the guidance in Reference 1 would be:

D (width of CB flange) = [ ].

d (OD of upper CB section) = [ ].

r (fillet radius) = [ ].

Therefore;

D/d = 1.123.

r/d = 0.014.

From Reference 1, Figure 73:

$K_t = 2.8$.

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.

References:

1. R.E. PETERSON: Stress Concentration Factors, 1974, Wiley-Interscience Publication.

ANP-10306P Technical Report Markups

4.2.2.3 *Modal Characterization*

Prior to the modal characterization test, static tests are performed on the HYDRAVIB mock-up to verify that the stiffness of the lower internal assembly and its restraint at the CB flange is modeled accurately.

03.09.02-77

This was accomplished through characterization tests with the hold-down ring to determine the stiffness for the one dimensional translational spring elements that were used with the FEM at the CB flange juncture. To confirm the spring stiffness's and verify the accurate behavior of this aspect of the numerical model, the bending stiffness of the lower internal assembly was assessed by laterally displacing the assembly at the LSP elevation to obtain the resulting displacement at this elevation from the numerical model. Considering the same loading scheme, the bending stiffness of the mockup was assessed by recording the applied load and resulting displacement at the LSP elevation. The stiffness values obtained through FEM calculation correlates closely with that obtained through the experimental observations.

After obtaining the numerically derived modal solution for the dry and wet modal frequencies (considering the hydrodynamic effects of the fluid), a comparison of the experimentally derived modal solution is performed as the bases for the validation of the numerical solution.

Applying an ideal "clamped boundary condition" at the CB flange location would represent an infinite stiffness of the translational spring elements, which would increase the modal frequencies of the CB lower assembly and result in a reduction in the response of the CB to turbulence. This conservatism in the type of boundary conditions was applied to both the scale and full scale analytical models.

Dry Modal Frequency Comparison

The comparison of the dry modal frequencies is provided in Table 4-1. These results demonstrate that the CB "pendulum" modal frequencies are similar using either the experimental or numerical approaches. Past experience has consistently shown that this mode is the most significant contributor to the motions of the internals.

It was observed during these tests that the modal frequencies of the HR were consistently overestimated by the numerical mode, which suggests that the HR model is stiffer than the