RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

APR1400 Design Certification

Korea Electric Power Corporation / Korea Hydro & Nuclear Power Co., LTD

Docket No. 52-046

RAI No.:	**151-8078**
SRP Section:	**3.9.2 – Dynamic Testing and Analysis of Systems Structures and Components**
Application Section:	**SRP 3.9.2**
Date of RAI Issue:	**08/10/2015**

Question No. 03.09.02-9

As a result of the audit conducted beginning on June 30, 2015, the staff identified additional detailed information that should be docketed to support the staff's safety finding associated with this section because Tier 2, Section 3.9.2.3.1 does not describe the hydrodynamic model or the method of calculating the forcing functions. In accordance with GDC 1 and 10 CFR 52.47, the applicant is requested to (1) describe how, within the hydrodynamic model, pump pulsation pressure fluctuation was translated to loads on RVI components; and (2) to clarify whether measured test data from one or four pumps were used and justify that the assumption of in-phase pressure fluctuation from four pumps operating is conservative.

Response

(1) The pump pulsation pressures for the RVI components are obtained using the test data for the following representative locations based on the valid prototype CVAP test data (Reference 1):

- RV Inlet Nozzle Location
- Control Element Assembly Guide Tube
- Incore Instrumentation Guide Tube
- Upper Guide Structure Support Plate
- Control Element Assembly Shroud

The RV inlet nozzle location data is used as input in solving the wave equation on the core support barrel (Reference 2). The other location data are also used for obtaining the pump pulsation pressures for the other RVI components as follows, which is described in Reference 2.

- Perpendicular pressure wave: when the pressure wave axis is perpendicular to the axis of the component
- Parallel pressure wave: when the pressure wave axis is parallel to the axis of the component.

The pump pulsation pressures for the RVI components are transferred as input to the structural response analysis. The overall procedure showing the hydraulic load analysis methodology is shown in Figure 3-1, Summary of Analytical Methodology of Reference 2.

(2) The measured data for all available conditions are taken from the valid prototype CVAP test data in accordance with the following steps:

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Accordingly, the assumption of in-phase pressure fluctuation from four pumps is conservative.

Used Measured Test Conditions for RV Inlet Location from Reference 1

Test Condition (PVMP #)	No. of Pumps
1	1
2	2
3	3
4	2
5	2
6	3
7	2
8	3
11	3
12	2
14	2
15	4

References

1. CEN-263(V)-P, Rev.1-P, A Comprehensive Vibration Assessment Program for Palo Verde Nuclear Generating Station Unit 1 (System 80 Prototype), Combustion Engineering, Inc., January 1985.
2. APR1400-Z-M-NR-14009-P, Rev.0, Comprehensive Vibration Assessment Program for the Reactor Vessel Internals, KEPCO & KHNP, November 2014.

Impact on DCD

There is no impact on the DCD.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Reports

There is no impact on any Technical, Topical or Environmental Reports.

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Question No. 03.09.02-10

DCD Tier 2, Section 3.9.2.3.1.1 states that the random hydraulic forcing function is developed by experimental methods and the forcing function is modified to reflect the flow rate and density differences based on an analytical expression found in Reference 45. However, the staff did not find an expression that is physically suitable to modify the random turbulent flow loadings represented by power spectrum density. In accordance with GDC 1 and 10 CFR 52.47, the applicant was requested to provide a description of the experimental methods and the analytical expression that modified the random forcing functions.

Response

The wording used in the DCD Tier 2, Section 3.9.2.3.1.2 was intended to indicate that the random hydraulic forcing function is developed based on the System 80 CVAP testing data. A DCD markup is attached to include a more accurate description. Reference 45 used in DCD Tier 2, Section 3.9.2.3.1.2 provides several expressions for the normalized power spectral density (PSD) which are of the following form:

$$\frac{G_p(f)}{\rho^2 V^3 D_H} = \overline{G_p}(F) = \text{a function of } F$$

where,

f = frequency
ρ = Fluid density
V = Velocity
D_H = Hydraulic equivalent diameter

$G_p(f)$ = PSD
$\overline{G_p}(F)$ = Normalized PSD
F = Dimensionless frequency $\left(=\frac{fD_H}{V}\right)$

The expressions show that $G_p(f)$ is related to the fluid properties (ρ and V) when the normalized PSD is restored to the original PSD. The forcing function is modified when required to reflect the flow rate (velocity) or density differences between the testing data and the design.

Impact on DCD

The DCD Tier 2, Section 3.9.2.3.1.2 will be revised as indicated in the Attachment.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Reports

There is no impact on any Technical, Topical or Environmental Reports.

3.9.2.3.1 Hydraulic Forcing Function

3.9.2.3.1.1 Deterministic Forcing Function

An analysis based on a hydrodynamic model is used to obtain the relationship between RCP pulsations in the inlet ducts and the deterministic pressure fluctuations on the core support barrel. A detailed description of this model and subsequent solution are given in References 41 and 42. The model represents the annulus of coolant between the core support barrel and the reactor vessel. In deriving the governing hydrodynamic differential equation for the model, the fluid is taken to be compressible and inviscid. Linearized versions of the equations of motion and continuity are used. The excitation on the hydraulic model is harmonic with the frequencies of excitation corresponding to pump rotational speeds and blade passing frequencies.

The dynamic force on the upper guide structure assembly is due to flow-induced forces on the tube bank. The deterministic components of these forces are caused by pressure pulsations at harmonics of the pump rotor and blade passing frequencies, and vortex shedding due to crossflow over the tubes.

The in-core instrumentation (ICI) nozzles and the skewed beam supports for the ICI support plate of the lower support structure are excited by deterministic and/or random, flow-induced forces. The deterministic component of this loading is due to pump-related pressure fluctuations and vortex shedding due to crossflow.

Data from the System 80 preoperational test (References 43 and 44) is used to determine the magnitude of these pulsations at the pump rotor and blade passing frequencies and their harmonics.

based on the System 80 preoperational test (Reference 44)

3.9.2.3.1.2 Random Forcing Function

The random hydraulic forcing function is developed ~~by experimental methods~~. The forcing function is represented in the form of power spectral density together with associated coherence area. The forcing function is modified to reflect the flow rate and density differences based on an analytical expression found in Reference 45.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

APR1400 Design Certification

Korea Electric Power Corporation / Korea Hydro & Nuclear Power Co., LTD

Docket No. 52-046

RAI No.:	**151-8078**
SRP Section:	**03.09.02 – Dynamic Testing and Analysis of Systems, Structures, and Components**
Application Section:	**3.9.2**
Date of RAI Issue:	**08/10/2015**

Question No. 03.09.02-11

DCD Tier 2, Section 3.9.2.4 states that evaluation of steam generator internals is included in Appendix A of the CVAP report, APR1400-Z-M-14009-P, Rev. 0. The staff reviewed Appendix A of this report and found that it presents an analysis based on the turbulent loading that determines that the stresses in the critical locations are small and acceptable. The analysis, however, does not address the fluid-structural interaction or flow-induced vibration due to cross flow conditions. In light of the recent operating experience with steam generator tube degradation at San Onofre Nuclear Generating Station, the applicant is requested to demonstrate that the APR1400 steam generator tube bundle design will prevent such degradation by (1) discussing the dynamic characteristics of the U-bend assembly including frequencies and mode shapes and describing the U-bend support configuration, or (2) provide the comparison between the APR1400 steam generators and similar steam generators (such as Palo Verde Nuclear Generation Station (PVNGS) Unit 1 replacement steam generators) that have operated without such adverse flow effects. The comparison should include geometry (size); numbers of horizontal and vertical supports and tube-to-support wear type; the steam generator design parameters as listed in APR1400 DCD Tier 2, Table 5.4.2-1; and operating conditions (steam quality, pressure, temperature and flow rates). This information is necessary for the staff to make a finding in accordance with 10 CFR 52.47(a)(22) that operating experience insights have been incorporated into the design.

Response

This response includes (1) the flow induced vibration (FIV) assessment of the APR1400 steam generator (SG) tube bundle including the modal analysis results, and (2) the comparison of the APR1400 SG and OPR1000 (Optimized Power Reactor 1000) SG tube bundle designs with the operating experience of the OPR1000 SGs. The OPR1000 SGs have been operating in Korean

Nuclear Power Plants without experiencing the adverse flow effects of fluid elastic instability (FEI) that abruptly occurred in the SONGS replacement steam generators (RSGs).

(1) FIV Assessment of APR1400 SG

There are four (4) regions in the APR1400 SG, where the secondary side flow crosses the tube bundle as shown in Figure 1-1. Three (3) of them are at the flow entrance regions and the other is at the flow exit region. While the flow at the entrance region is either sub-cooled (at the economizer) or saturated liquid (at the evaporator), the flow at the exit region is two-phase.

Prior to discussing the dynamic characteristics of the U-bend assembly, the geometry of tube and tube support configuration is presented. Figure 1-2 shows the general arrangement of the tube supports for the APR1400 SG. Detail-A and Detail-B in Figure 1-2 show the assembled tube and tube support configuration at the straight and U-bend tube regions, respectively. The abbreviation used in Figure 1-2 is Half Eggcrate (HEC), Full Eggcrate (FEC), Partial Eggcrate (PEC), Diagonal Tube Support Bar (DTSB), Vertical Tube Support Bar (VTSB), and FDP (Flow Distribution Plate). In the straight tube region, the tubes are arrayed in a triangular pattern with 1.00 inch pitch. In the U-bend tube region, the tubes are arranged in a rotated square pattern with 1.231 inch pitch. A total of 13,102 tubes (175 rows and 207 lines) are installed in an APR1400 SG. Two shapes of U-bend tubes are used; tubes in row numbers 1 through 17 are 180^o bend and tubes in rows 18 through 175 are double 90^o bends (so called square bends) as shown in Figure 1-3. Figure 1-4 shows the detailed view of the tube supports at the U-bend tube region (the tubes are omitted for clarity).

For the straight region of tube, the eggcrates restrict the tube motion in the horizontal direction.

For the U-bend region of tube, the VTSBs and DTSBs suppress the out-of-plane tube motion. The horizontal strips inserted into the VTSB (See Figure 1-2 and Figure 1-4), which are installed between the double 90^o bend tube rows, restrain the in-plane tube motion. The usage of horizontal strips is the key difference in the U-bend tube support design from other SG U-bend tube support designs that use only the U-bend shape tubes.

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Figure 1-1 Cross Flow Regions in APR1400 SG

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Figure 1-2 General Arrangement of Tube Supports

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Figure 1-3 Shapes of U-bend Tubes

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Figure 1-4 Tube Supports at U-bend Region

The FIV evaluation for the APR1400 SG tube bundle is performed for the tube rows in the flow exit region which have the longest unsupported span, including the last tube row 175. Since each tube row has several or dozens of tube lines, the specific tube line is selected based on the dynamic pressure of the secondary side flow obtained from the ATHOS3 thermal-hydraulic analysis.

In the flow entrance region, the straight portion of tube up to the third full eggcrate (FEC) is investigated since the most susceptible flow entrance region to the FIV is beneath the first FEC as shown in Figure-1 and Figure-2.

Table 1-1 shows the row and line number of selected tubes in the flow exit region.

The ANSYS computer program (DCD Tier 2, Section 3.9.10, Reference 10) is used to model the tube and analyze the modal response of the tube. The element types of [ ]TS are used.

Figure 1-5 shows the finite element model (FEM) for the tube modal analysis in the flow exit region. The mode shapes from the 1st to the 10th modes of tube [ ]TS are represented in Figure 1-6. Table 1-2 summarizes the tube frequencies for flow the exit region. The mode shapes from the 1st to the 10th modes of tubes of interest are listed in Table 1-3. Figure 1-7 shows the FEM for the tube modal analysis in the flow entrance region. The FEM for the feedwater flow entrance region is identical to that for the cold side recirculating flow entrance region. The mode shapes from the 1st to the 30th mode of cold side flow entrance are presented in Figure 1-8. Table 1-4 summarizes the tube frequencies for the flow entrance region.

Fluid Elastic Instability (FEI) and Random Turbulence Excitation (RTE) are evaluated for the APR1400 SG tubes. For the FEI evaluation, instead of using a stability ratio (SR) of 1.0, which predicts the onset of FEI, a [ ]TS is conservatively considered as a design goal for the SR. The SR is the effective velocity (V_{eff}) over the critical velocity (V_{cr}). For the RTE evaluation, the design goal for the root mean square (RMS) displacement is [ ]TS.

The FEI evaluation results for the flow exit and flow entrance regions are tabulated in Table 1-5 and Table 1-6, respectively. The RTE evaluation results for the flow exit and flow entrance regions are summarized in Table 1-7 and Table 1-8, respectively.

Based on the FEI and RTE evaluation, the [ ]TS tube at the flow exit region is found to be the most important tube showing the highest SR of [ ]TS and the maximum RMS displacement of [ ]TS. The summary of the FIV evaluation results for the APR1400 SG tube bundle is documented in KEPCO/KHNP Report No. APR1400-H-N-NR-14002-P, “Summary Stress Report for Steam Generator.”

Table 1-1 Tubes for FIV Assessment in Flow Exit Region

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Table 1-2 Tube Frequencies for Flow Exit Region

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Table 1-3 Tube Mode Shapes for Flow Exit Region

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Table 1-4 Tube Frequencies for Flow Entrance Region

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Table 1-5 FEI Evaluation of Flow Exit Region

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Table 1-5 FEI Evaluation of Flow Exit Region (Cont'd)

TS

Table 1-5 FEI Evaluation of Flow Exit Region (Cont'd)

TS

Table 1-5 FEI Evaluation of Flow Exit Region (Cont'd)

TS

Table 1-6 FEI Evaluation of Flow Entrance Region

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Table 1-7 RTE Evaluation of Flow Exit Region

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Table 1-8 RTE Evaluation for Flow Entrance Region

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Figure 1-5 FEM for Tube Modal Analysis in Flow Exit Region

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Figure 1-5 FEM for Tube Modal Analysis in Flow Exit Region (Cont'd)

TS

Figure 1-5 FEM for Tube Modal Analysis in Flow Exit Region (Cont'd)

TS

Figure 1-6 Mode Shapes from 1st to 10th Mode of Tube R23L128

TS

Figure 1-6 Mode Shapes from 1st to 10th Mode of Tube R23L128 (Cont'd)

TS

Figure 1-7 FEM for Tube Modal Analysis of Flow Entrance Region

TS

Figure 1-8 Mode Shapes from 1st to 30th Mode of Cold Side Flow Entrance

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Figure 1-8 Mode Shapes from 1st to 30th Mode of Cold Side Flow Entrance (Cont'd)

TS

Figure 1-8 Mode Shapes from 1st to 30th Mode of Cold Side Flow Entrance (Cont'd)

TS

Figure 1-8 Mode Shapes from 1st to 30th Mode of Cold Side Flow Entrance (Cont'd)

TS

Figure 1-8 Mode Shapes from 1st to 30th Mode of Cold Side Flow Entrance (Cont'd)

(2) Comparison of APR1400 and OPR1000 SG and Operating Experience of OPR1000 SG

The design of the APR1400 SG is compared to that of the OPR1000 SG, which has operated in the Korean Nuclear Power Plants without experiencing such an abrupt FEI mechanism as that which was observed in the SONGS Units 2 & 3 RSGs.

The design features of the tube and tube supports in the APR1400 SGs and OPR1000 SGs are identical in physical aspects such as tube outer diameter, tube wall thickness, tube pitches and arrays in the straight and U-bend region, configuration of tube and tube support assembly as shown in Figure 1-2 through Figure 1-4. The only major difference is the tube support spacing as shown in Figure 2-1. To increase the heat transfer area from the OPR1000 SG to the APR1400 SG, the tube bundle is widened instead of lengthened. Thus, the overall heights of the tube bundles are similar. More eggcrates and VTSBs are installed in the APR1400 SG, thus the tube support spacing is substantially reduced compared to the OPR1000 SG. Table 2-1 compares the design parameters between OPR1000 SG and APR1400 SG.

Table 2-2 provides the operating experience, tube wear status, of the OPR1000 SG (Han-ul Units 5 & 6). These are the [ ]^{TS} In-service Inspection (ISI) results performed in [ ]^{TS} for Unit 5 and [ ]^{TS} for Unit 6. Han-ul Units 5 & 6 started the commercial operation in July 2004 and April 2005, respectively. There are a few tube-to-support wear indications in both units; however, there is no tube-to-tube wear which was reported as a result of in-plane FEI in the SONGS Units 2 & 3 RSGs. This operating experience is further justification on the soundness of the APR1400 SG tube and tube support design.

Table 2-1 Comparison of Tube Bundle Design of OPR1000 SG and APR1400 SG

Parameter	OPR1000 SG	APR1400 SG
Tube Material	Alloy 690 Thermally Treated	
Tube Outer Diameter [in]	0.75	
Tube Wall Thickness [in]	0.042	
Tube Pitch [in] (Straight)	1.0 (Triangular)	
Tube Pitch [in] (U-Bend)	1.231 (Rotated Square)	
Number of Tubes Installed [ea]	[ ]TS	13102
Row Number x Line Number	[ ]TS	175 x 207
Tube Shape - 180^{o} Bend - Double 90^{o} Bend	 Row [ ]TS Row [ ]TS	 Row 1 through 17 Row 18 through 175
Average Active (Heated) Tube Length [ft]	[ ]TS	63.62
Heat Transfer Area per SG [ft^{2}]	[ ]TS	163670
Overall Height of Tube Bundle [in]	[ ]TS	[ ]TS
Height of Straight Tube [in]	[ ]TS	[ ]TS
Height of U-Bend Tube [in]	[ ]TS	[ ]TS
Heat Transfer Rate per SG [10^{6} Btu/hr]	[ ]TS	6830
Steam Pressure at Steam Dome [psi]	[ ]TS	1000
Steam Mass Flow Rate [10^{6} lb/hr]	[ ]TS	8.975
Steam Mass Quality [%]	99.75	

Table 2-2 Tube Wear Status[1] of Han-ul Units 5 & 6

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Figure 2-1 Comparison of Tube Support Spacing between OPR1000 and APR1400 SGs

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Impact on DCD

There is no impact on the DCD.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Reports

There is no impact on any Technical, Topical, or Environmental Reports.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

APR1400 Design Certification

Korea Electric Power Corporation / Korea Hydro & Nuclear Power Co., LTD

Docket No. 52-046

RAI No.:	**151-8078**
SRP Section:	**3.9.2 - Dynamic Testing and Analysis of Systems Structures and Components**
Application Section:	**Section 3.9.2**
Date of RAI Issue:	**08/10/2015**

Question No. 03.09.02-12

APR1400-Z-M-14009-P, Appendix B, "Vibration Assessment for the Reactor Coolant System Piping and the Piping Attached to the Steam Generator," states that the vibration assessment program consists of a vibration and stress analysis program, and a flow excited acoustic resonance measuring and inspection program are performed for the following system piping: (1) reactor coolant system (RCS) piping, (2) main steam system (MS) piping, (3) feedwater system (FW) piping, and (4) condensate system (CD) piping, in conformance with RG 1.20 and SRP Section 3.9.2. The applicant is requested to justify not including vibration assessment for the shutdown cooling and other emergency core cooling system lines, given operating experience at the similar PVNGS plant where a flow-excited acoustic resonance was experienced in the shutdown cooling system, resulting in leaking and failure of an isolation valve. This information is necessary for the staff to make a finding in accordance with 10 CFR 52.47(a)(22) that operating experience insights have been incorporated into the design.

Response

Vibration assessment for the shutdown cooling system lines is not required for the APR1400 based on the operating experience insights from the PVNGS plant and the OPR1000.

Palo Verde Unit 1 experienced extensive outages or periods of low power operation during the first half of 2006 due to excessive vibration levels in the Unit 1 Train A Shutdown Cooling System (SCS) suction line. Arizona Public Service Company (APS) conducted extensive investigations to determine the source of the SCS suction line vibrations and to determine the reasons for the increased vibration levels. APS concluded that the vibration was flow induced and was caused by coupling between an excitation source, vortex shedding in the SCS suction line tee due to RCS flow over the SCS suction nozzle, and an acoustic resonator. After evaluating many options, APS resolved the problem by moving the SCS suction line isolation valve SI-V651 nearer to the RCS hot leg. The new location of the SCS suction line isolation valve SI-V651 is 11 feet from the RCS nozzle compared to the original location which was approximately 52.5 feet from the nozzle.

The SCS suction line designs of the OPR1000 and APR1400 are similar to the PVNGS plant SCS suction line re-design. Specifically, the SCS suction line diameters (16 inches) and hot leg diameters (42 inches) of the OPR1000 and APR1400 are the same as those of the PVNGS plant. The locations of the SCS suction line isolation valves SI-V651/V652 are 11 feet 4 inches from the RCS nozzle for the OPR1000 and 12 feet 8 inches from the RCS nozzle for the APR1400. All locations are similar to the new location of SI-V651 in the PVNGS plant. In addition, an excessive vibration in the SCS suction line has not been reported in any of the OPR1000 plants. Therefore, based on the design configuration and similar plant operating experience, the possibility of excessive vibration levels in the SCS suction line experienced in the PVNGS plant Unit 1 SCS Train A is very low for the APR1400 and inclusion in the vibration assessment program is not necessary.

Impact on DCD

There is no impact on the DCD.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on any Technical Specifications.

Impact on Technical/Topical/Environmental Reports

There is no impact on any Technical, Topical or Environmental Reports.

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Application Section:	**3.9.2**
Date of RAI Issue:	**08/10/2015**

Question No. 03.09.02-13

DCD Tier 2, Section 3.9.2.5.1 states that the seismic dynamic analysis of the reactor internals including the core is performed separately for the horizontal and vertical directions. The nonlinear horizontal model, as shown in DCD Tier 2, Figure 3.9-16, was constructed to consider gaps between internal components (between core and core shroud, and between core support barrel and reactor vessel) and the large relative displacements occurred during the SSE event. The vertical non-linear model (DCD Tier 2, Figure 3.9-18) was constructed considering the possibility of the core assembly lifting off the support plate. The applicant further states that the mathematical model also includes hydrodynamic effects. However, the staff did not find that Section 3.9.2.5 provided information regarding fluid-structural interactions for the models shown in Figures 3.9-16 and 3.9-18.

SRP acceptance criterion II.5.D states that the effects of flow upon the mass and flexibility properties of the system should be addressed. Therefore, the applicant is requested to provide information as to how fluid-structural interaction effects are accounted for in the mass and flexibilities of reactor internals as part of the dynamic modeling.

Response

The effect of fluid-structure interaction (FSI) between reactor internals structures is characterized by a hydrodynamic mass matrix. The hydrodynamic mass matrix consists of off-diagonal hydrodynamic coupling terms and diagonal hydrodynamic added mass terms. The theory and methodology for hydrodynamic mass matrix are presented in detail in Section 7, CENPD-178 (Reference 1).

The hydrodynamic mass theory is incorporated into the CESHOCK computer program. The CESHOCK computer program accounts for the FSI effect between two adjacent structures, separated by a fluid-filled gap, by applying a hydrodynamic mass matrix to evaluate the fluid forces on the motion of structures.

Major components of the APR1400 reactor vessel internals (RVI) consist of concentric cylindrical structures separated by an annulus for coolant flow. To account for the FSI effect between the structures, the RVI horizontal model includes the hydrodynamic coupling and added mass between nodes on the RV and CSB, the CSB and CS, the CSB and UGS, and the UGS and IBA (see Figure 3-3, Reference 2). The hydrodynamic coupling mass and added mass terms are determined from the annulus geometry between structures.

In vertical direction, there is no structure separated by a fluid-filled gap that causes a considerable hydrodynamic effect, and many flow holes in the UGSSP, FAP and ICI nozzle support plate mitigate the FSI effect. Therefore, the FSI effect is not considered in the RVI vertical model.

Notes:

CSB: core support barrel	CS: core shroud	FAP: fuel alignment plate
IBA: inner barrel assembly	ICI: in-core instrumentation	RV: reactor vessel
UGS: upper guide structure	UGSSP: upper guide structure support plate	

References

1. CENPD-178, “Structural Analysis of Fuel Assemblies for Seismic and Loss of Coolant Accident Loading,” Combustion Engineering, Inc., Rev. 1, August 1981.

2. APR1400-Z-M-NR-14010-P, "Structural Analysis of Fuel Assemblies for Seismic and Loss of Coolant Accident Loading", Rev. 0, KHNP, November 2014.

Impact on DCD

There is no impact on the DCD.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Reports

There is no impact on any Technical, Topical or Environmental Reports.

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Question No. 03.09.02-14

DCD Tier 2, Section 3.9.2.5.1 states that the input excitation to the internals model is the response time-history of the reactor vessel at the internals support determined from the RCS analysis. Coupling effects between the internals and reactor vessel are accounted for by including a simplified representation of the internals with the RCS model. In addition, the applicant stated that the nonlinear seismic response and impact forces for the internals and fuel are determined using the CESHOCK computer program, which is described in the DCD Tier 2, Section 3.9.1. The input excitation for the model is the time-history acceleration of the reactor vessel. The procedures used to account for damping in the analysis of the reactor internals and core are provided in DCD Tier 2, Section 3.7.2.14.

Therefore, the applicant is requested to provide justification for using the procedures for analysis of damping provided in DCD Tier 2, Section 3.7.2.14 (which are for the modal analysis of a linear structural system or the proportional viscous damping using direct integration method for a linear system) also for the non-linear dynamic analysis described in DCD Tier 2, Section 3.9.2.5.

Response

Reactor vessel internals (RVI) and core seismic analyses are performed with a nonlinear time history analysis using the CESHOCK computer program, which solves the differential equation of motion for a multi-degree-of-freedom system using direct integration method. To account for damping in the seismic analysis of the RVI and core, the proportional damping matrix provided in DCD Tier 2, Section 3.7.2.14 is used, which has the form as follows:

$[C] = \alpha\,[M] + \beta\,[K]$

Where:
[C] = damping matrix, [M] = mass matrix, [K] = stiffness matrix
α, β = constants

The damping ratio ξ_i at any frequency f_i is expressed as follows:

$$\xi_i = \frac{\alpha}{4\pi f_i} + \beta\pi f_i$$

The constants α and β are computed by specifying the damping ratio at two frequencies f_1 and f_2.

The APR1400 RVI components have major natural frequencies between [ ]TS which are determined from the modal analysis results of the RVI models under various gap conditions to consider the non-linear effect of the gap between structures. Based on the above frequencies, the lower and upper bound frequencies [ ]TS are conservatively selected so that all significant modes of the RVI components lie between the frequencies, f_1 and f_2. The damping value of the RVI components for SSE is 4 percent of the critical damping in accordance with RG 1.61.

Figure 1 shows that damping values for all frequencies between [ ]TS are lower than 4 percent of the critical damping.

TS

[ ]

Figure 1. Damping Value for RVI components

For the fuel assemblies, the values of α and β are calculated by the fuel assembly natural frequencies at significant modes and their critical damping ratios which are determined from the fuel assembly lateral vibration test (see Section 5.2.2, Reference 1).

Reference

1. APR1400-Z-M-NR-14010-P, "Structural Analysis of Fuel Assemblies for Seismic and Loss of Coolant Accident Loading", Rev. 0, KHNP, November 2014.

Impact on DCD

There is no impact on the DCD.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

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Impact on Technical/Topical/Environmental Reports

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Date of RAI Issue:	**08/10/2015**

Question No. 03.09.02-15

GDC 1 relates to the design, fabrication, erection, and testing of SSCs in accordance with quality standards commensurate with the importance of the safety function to be performed. To enable the staff to make a conclusion as to the APR1400 design's compliance with GDC 1, specific to the appropriate correlation of tests and analyses of reactor internals, the applicant is requested to provide the following information:

- Comparison of the measured response frequencies with the analytically obtained natural frequencies of the reactor internals for validation of the mathematical models used in the analysis
- Comparison of the analytically obtained mode shapes with the shape of measured motion for identification of the modal combination or verification of a specific mode
- Comparison of the response amplitude time variation and the frequency content from test and analysis for verification of the postulated forcing function
- Comparison of the maximum responses from test and analysis for verification of stress levels
- Comparison of the mathematical model for dynamic system analysis under operational flow transients and under combined LOCA and SSE loadings between APR1400 and valid prototype plant.
- Comparison of measurements and predictions of any adverse flow phenomena (e.g., flow excited acoustic and/or structural resonances) for validation of the model(s) predicting the loading induced by the phenomena

Response

- A comparison of the measured response frequencies with the analytically obtained natural frequencies of the reactor vessel internals (RVI) is provided in the following table and in Table 2-4 of APR1400-Z-M-NR-14009-P, Rev. 0 [Reference 1].

TS

- A comparison of the analytically obtained mode shapes with the shape of measured motion for identification of the modal combination or verification of a specific mode is provided as follows:

  The predicted mode shapes of the valid prototype were verified through the measurement during the comprehensive vibration assessment program (CVAP) included in Reference 2. However, only the first or second mode of the measurement was identifiable. The detailed deflected shapes were predicted since strong turbulence existed in the low frequency range and the number of instruments used and their signals were limited for determining detailed mode shapes. The predicted mode shapes of the APR1400 RVI are compared with those of the valid prototype as follows:

1) Core Support Barrel (CSB)

TS

2) Upper Guide Structure (UGS)

TS

3) Lower Support Structure (LSS) / ICI (In-Core Instrumentation)

TS

- A comparison of the response amplitude time variation and the frequency content from test and analysis for verification of the postulated forcing function is provided as follows:

  Comparison of the pump pulsation pressures is provided for the following components.

  - Control Element Assembly Guide Tube (CEA GT)
  - In-Core Instrumentation Guide Tube (ICI GT)
  - Upper Guide Structure Support Plate (UGSSP)
  - Control Element Assembly Shroud (CEA Shroud)

<u>Measured RMS Pressure (4-Pump), psi (Valid Prototype)</u>

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Used RMS Pressure (4-Pump), psi (APR1400)

TS

It is noted that the RMS pressures are used as inputs to calculate the pump-induced pressure loads across the components as described in Reference 1.

- A comparison of the maximum stresses as a result of the test and analysis is provided in the following table [see Table 2-5 and Table 3-26 of Reference 1].

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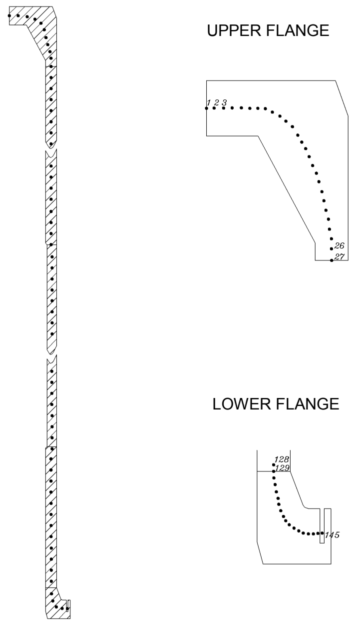
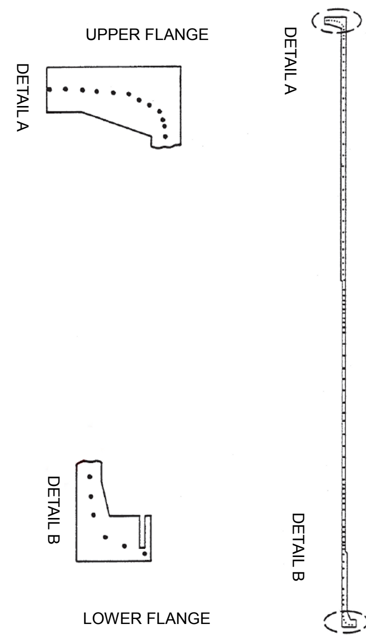
- A comparison of the mathematical model for dynamic system analysis under operational flow transients and under combined LOCA and SSE loadings between APR1400 and valid prototype plant is provided as follows:

1) CSB Model

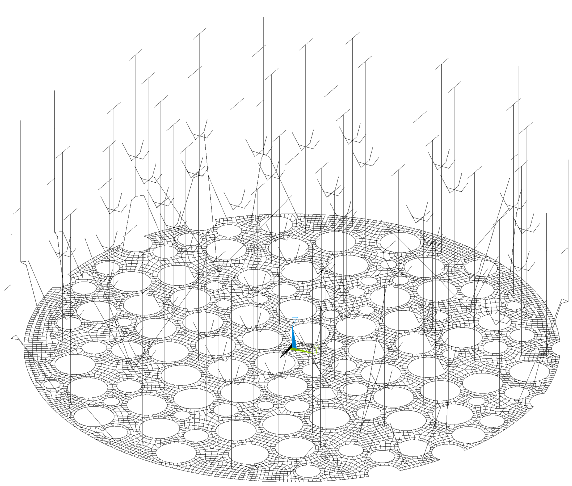


APR1400
[see Figure 3.9-3 of DCD Tier 2]

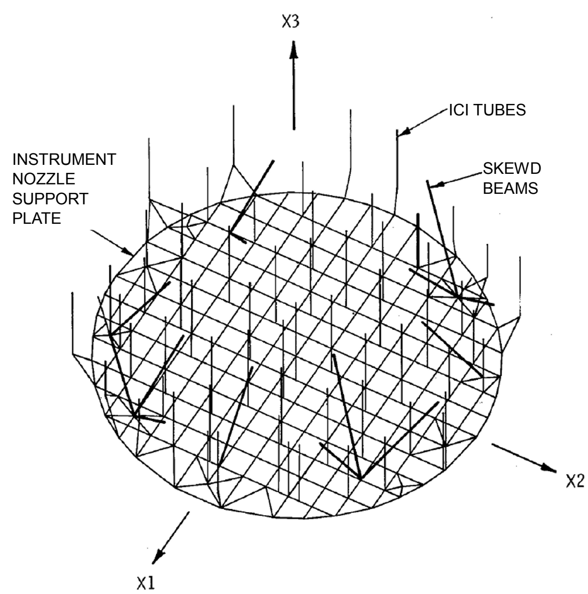


Valid Prototype
[see Figure 5.2-2 of Reference 3]

2) LSS/ICI Model

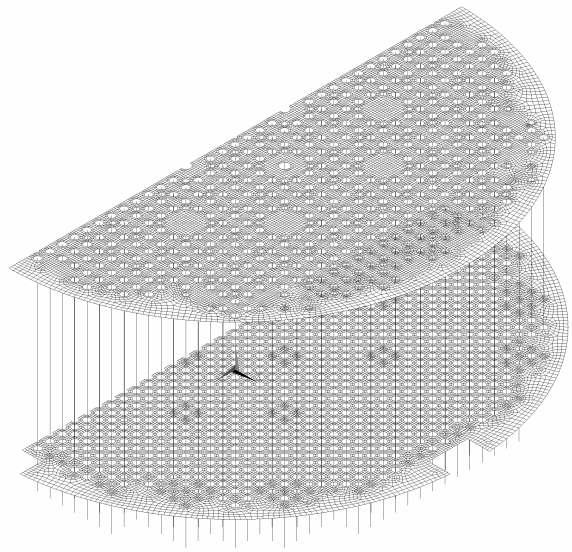


APR1400
[see Figure 3.9-6 of DCD Tier 2]

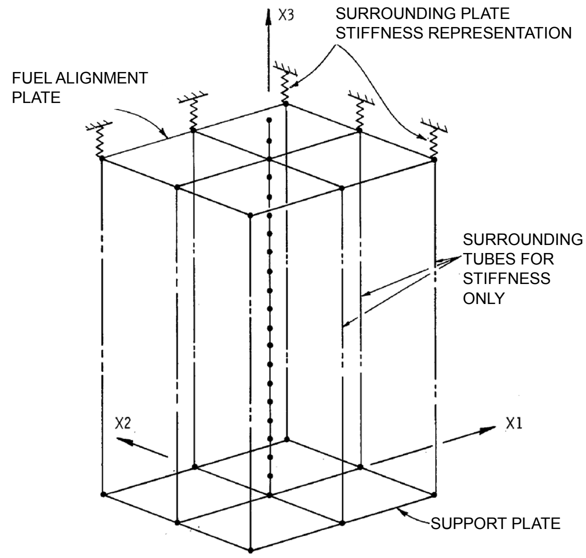


Valid Prototype
[see Figure 3.9.2-4 of Reference 3]

3) CEA Guide Tubes Model

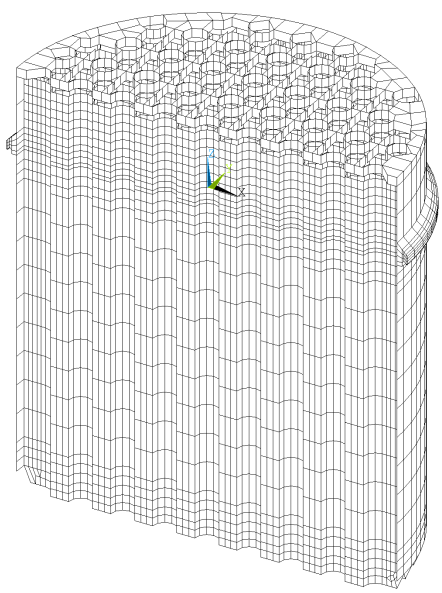


APR1400
[see Figure 3.9-5 of DCD Tier 2]

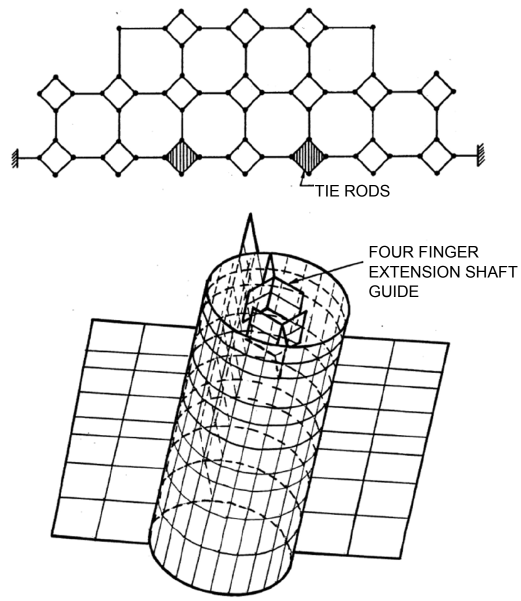


Valid Prototype
[see Figure 5.3-1 of Reference 3]

4) Inner Barrel Assembly (IBA) / CEA Shroud Assembly Model

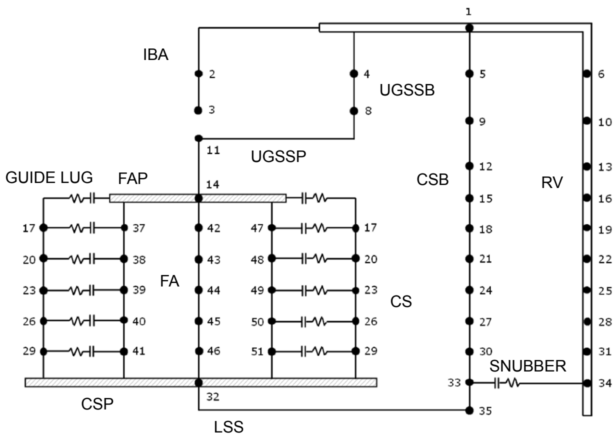


Inner Barrel Assembly of the APR1400
[see Figure 3.9-4 of DCD Tier 2]

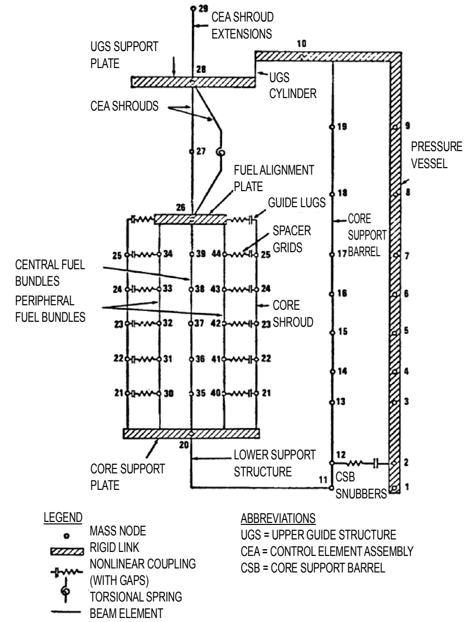


CEA Shroud Assembly of the Valid Prototype
[see Figures 4.2-6 and 4.2-9 of Reference 4]

5) Reactor Internals Horizontal Model for Dynamic Analysis

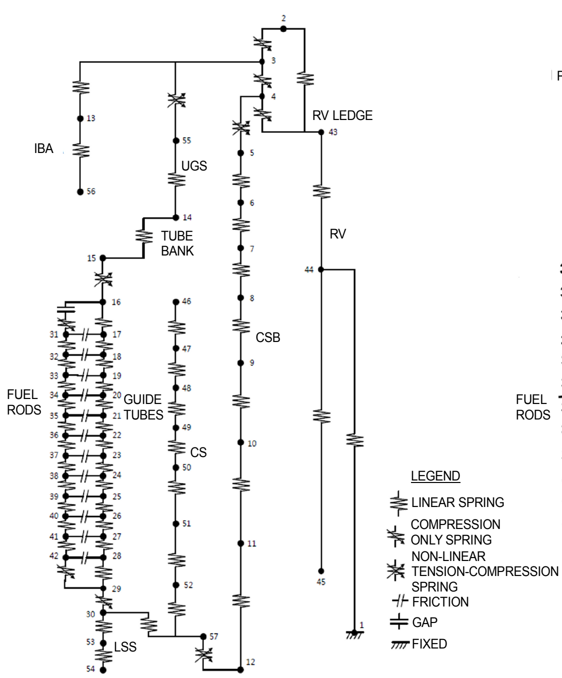


APR1400
[see Figure 3.9-16 of DCD Tier 2]

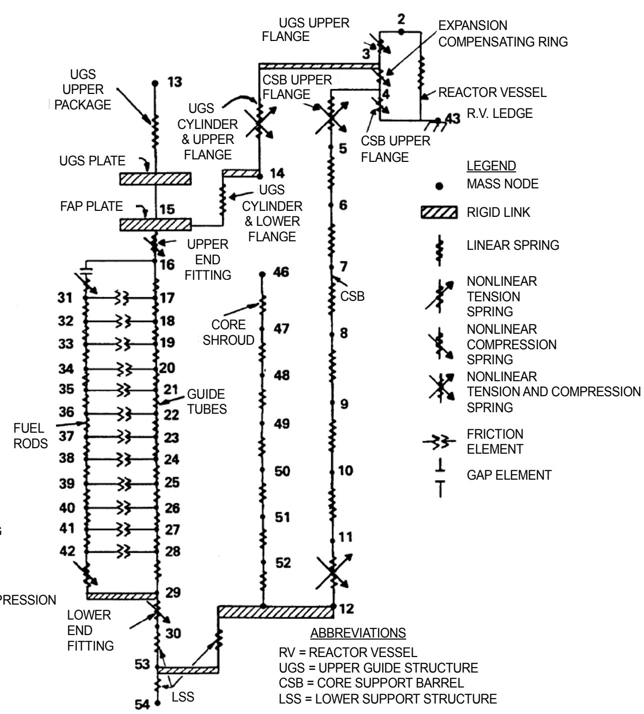


Valid Prototype
[see Figure 3-4 of Reference 5]

6) Reactor Internals Vertical Model for Dynamic Analysis



APR1400
[see Figure 3.9-18 of DCD Tier 2]



Valid Prototype
[see Figure 3-13 of Reference 5]

- A comparison of the measurements and predictions of any adverse flow phenomena (e.g., flow excited acoustic and/or structural resonances) for validation of the model(s) predicting the loading induced by the phenomena is provided as follows:

  The flow induced loadings applied to the APR1400 RVI are based on the CVAP results of the valid prototype. Since the measured data was obtained under real circumstances during various valid prototype CVAP conditions and used to determine the predicted loadings for the structural response of the APR1400 RVI, the predicted loadings consider the effects from all phenomena within the reactor vessel including adverse flow phenomena. Therefore the forcing function includes adverse flow phenomena and their effects.

  The structural responses to the loadings for the APR1400 RVI are predicted to be much greater than the measurements of the valid prototype as shown in the fourth item above. The prediction includes the variations of forcing frequencies accounting for the uncertainties of the mathematical models and loadings including adverse flow phenomena.

References

1. APR1400-Z-M-NR-14009-P, Rev.0, Comprehensive Vibration Assessment Program for the Reactor Vessel Internals, KEPCO & KHNP, November 2014.
2. CEN-263(V)-P, Rev.1-P, A Comprehensive Vibration Assessment Program for Palo Verde Nuclear Generating Station Unit 1 (System 80 Prototype), Combustion Engineering, Inc., January 1985.
3. CEN-202, A Comprehensive Vibration Assessment Program for System 80 Reactor Internals Palo Verde Nuclear Generating Station Unit 1, Combustion Engineering, Inc.
4. CEN-267(V)-P, Rev.1-P, Final Report on the Performance Evaluation of the Palo Verde Control Element Assembly Shroud, Combustion Engineering, Inc., August 1984.
5. CENPD-178, Rev.1-P, Structural Analysis of Fuel Assemblies for Seismic and Loss of Coolant Accident Loading, Combustion Engineering, Inc., August 1981.

Impact on DCD

There is no impact on the DCD.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Reports

There is no impact on any Technical, Topical or Environmental Reports.