SUPPLEMENTAL 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**
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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 180o bend and tubes in rows 18 through 175 are double 90o 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 90o 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)

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

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

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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)

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

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

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

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

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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)

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

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

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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° Bend - Double 90° 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

Supplemental Response

The vibration test that was conducted to obtain the damping ratio used the real-sized steam generator tubes and tube supports, which were identical to those of the APR1400 and OPR1000 steam generator designs [Reference 1].

The test method was to excite the tubes with the random forces over the frequency of interest. The damping ratio was obtained from the specified frequency peaks in the frequency response spectra by means of the half power point bandwidth method. Various test cases were considered according to the tube and support configuration, test environment and the amplitude of the excitation forces. Figure 3-1 shows the schematics of the test bed layout.

The damping ratio acquired from the test was reviewed by comparing the results with other references; not only for assuring the adequacy of test data, but also to apply the damping ratio to the design. The other references include former experimental data, semi-empirical formulation in research papers, the ASME B&PV Code, and the safety evaluation/review guidelines suggested by the Korea Institute of Nuclear Safety (KINS).

Based on the test results and the review of the state-of-the-art references, the low bound fitting curve for damping ratio as a function of frequency was developed, which is shown in Figure 3-2.

Reference 1. Steam Generator Tube Vibration Tube Vibration Test, Y1201A-160TD-2001, Rev.0, Doosan Heavy Industries and Construction (Proprietary).

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Figure 3-1 Schematics of Test Bed Layout

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Figure 3-2 Low Bound Fit of Damping Ratio

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.