Transactions of the Korean Nuclear Society Spring Meeting
Jeju, Korea, May 22, 2009

Hydraulic and Structural Analysis Methodology of RVI CVAP in Shin-Kori 4

Kyu-Hyung Kim*, Do-Young Ko, and Yong-Soo Kim
Nuclear Engineering & Technology Institute, 25-1 Jang-dong, Yuseong-gu, Daejeon, 305-343, Republic of Korea
**Corresponding author: khkim123@khnp.co.kr*

1. Introduction

The NRC Regulatory Guide 1.20 requires verification of the structural integrity of the reactor vessel internals (RVI) against flow induced vibrations during normal operation and transient conditions [1]. The category of RVI is classified according to the similarity of RVI design and operating parameters with those of the reference plant.

Since the design of the CEA shroud, the thermal power, and RCS flow rate have changed from the reference plant, Palo Verde 1, and since Shin-Kori 3 and 4 are the first commercial plants of APR1400 design, we will verify the integrity of the design and manufacturing of RVI by performing the non-prototype category II CVAP (Comprehensive Vibration Assessment Program), consisting of analysis, limited measurements and inspection, as shown in Fig. 1. This paper describes the methodology of the analysis program for the RVI of Shin-Kori 4.

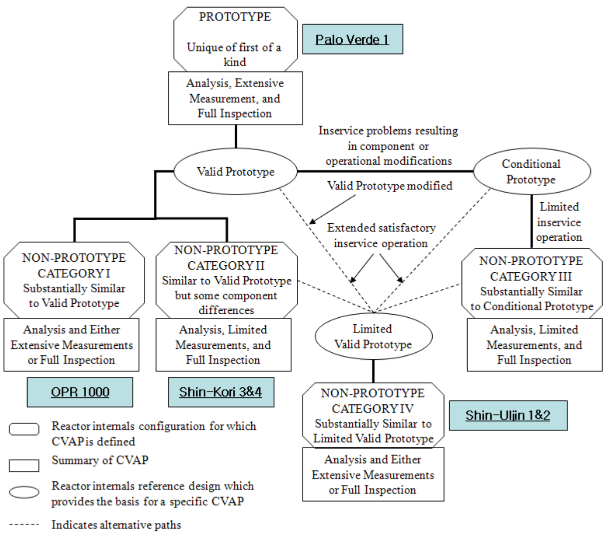

Fig. 1. Category of CVAP

2. RVI CVAP Experiences

2.1 Palo Verde 1

Because the reactor internals of Palo Verde 1 have a completely different concept from previous CE type nuclear power plants such as Main Yankee and Fort Calhoun, in terms of arrangement, design, size, and operation conditions, the reactor of Palo Verde 1 was categorized as a prototype, and the CVAP was performed as follows.

- Measurement of pressure, acceleration, strain, and displacement at 38 locations during pre-core HFT
- Because of the defects in the CEA shroud, CEA guide fingers were removed to minimize the stress concentration in the CEA, and four lateral snubbers in the shroud assembly were added to decrease the dynamic movement of the CEA shroud
- Verification of the RVI integrity for flow induced vibration according to analysis and testing of the modified CEA shroud [2].

2.2 Yonggwang 3 and 4

Because the CEA shroud of Yonggwang (YGN) 3 and 4 was changed to solve the crack problem detected in Palo Verde 1 and the thermal power was reduced from the reference plant, Palo Verde 1, in addition to vibration analysis, limited measurements were performed during the pre-core HFT of YGN 4. These measurements showed that the abnormal damage of RVI would not be generated over the plant's lifetime [3].

2.3 OPR1000

In OPR1000 (Optimized Power Reactor 1000) plants, the plants that followed YGN 3 and 4, the non-prototype category I RVI CVAP was performed with Palo Verde 1 as the reference plant. The RVI integrity was confirmed by the inspection before and after loading over a 10^6 cycle vibration during the pre-core HFT [4].

3. Shin-Kori 4 RVI CVAP plan

We planned to verify the integrity of RVI by performing the non-prototype category II CVAP in Shin-Kori 4 because the RVI designs of Shin-Kori 3 and 4 are different from that of the reference plant, Palo Verde 1, as follows:

- The type of CEA shroud is changed from assembled shroud to integrated shroud to modify the load path, in other words, the inner barrel assembly(IBA) is introduced
  - The shroud tube and web assembly are connected to an external cylinder
  - The tube, web, and cylinder assembly is supported by the UGS (Upper Guide Structure) upper flange
  - The tie rod is removed

- An increase in thermal power and coolant flow rate
- An increase in the thickness of the core support barrel (CSB) flange and fuel alignment plat
- An increase in the number of CEA guide tubes

The vibration in the UGS of Shin-Kori 4 will be mainly measured since the design of the CEA shroud is changed as mentioned above. According to the RVI CVAP experiences in Palo Verde and YGN 3 and 4, the analysis program of the RVI CVAP in Shin-Kori 4 will be planed and performed as follows.

4. Analysis procedure of RVI CVAP in Shin-Kori 4

The goal of the analysis program is to provide theoretical verification and a basis for the choice of components and areas to be monitored in the measurement and inspection programs. Analytical methodology, which is calculating hydraulic load, structural analysis, and structural response, is used to calculate the responsive stress of the reactor internals against flow induced vibrations (Fig. 2). The core support barrel, lower support structure, and upper guide structure in reactor internal components are analyzed.

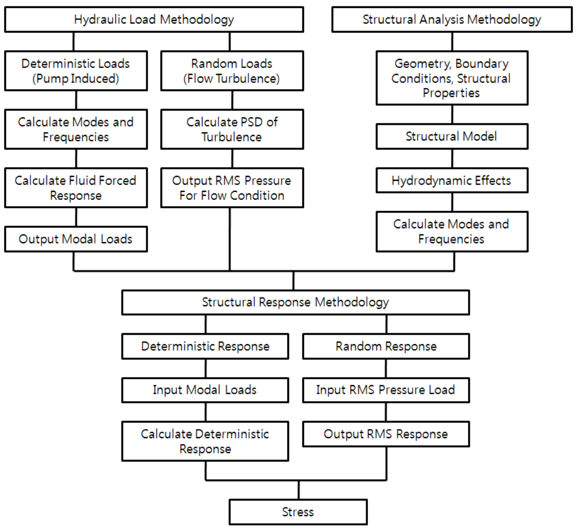

Fig. 2 Summary of analytical methodology

4.1 Hydraulic forcing function

Induced hydraulic forcing functions during steady and transient operation consist of deterministic hydraulic loads, which are related to pump pulsation and vortex shedding, and random hydraulic loads. Pump pulsation loads are due to the harmonic variations of the fluid pressure caused by reactor coolant pumps. Deterministic hydraulic loads caused by vortex shedding are a function of local cross flow velocities, the geometry and orientation of the structure, the intensity of the turbulence, and Reynolds Number. Flow induced random loads are primarily due to flow turbulence.

4.2 Structural analysis

A variety of FEMs (Finite Element Models) or multiple degree-of-freedom lumped mass models are used to calculate the natural frequencies and mode shapes of the reactor internals. The choice of particular method depends on the complexity of the structure and the nature of the hydraulic load. Currently, the fluid effect is calculated through solving the coupled fluid-structure problem or using hydrodynamic added mass.

The dynamic response of reactor internals due to deterministic and random forcing functions is calculated using the mode superposition. In either case, the response is a function of the magnitude of each load, its frequency and spatial distribution, and the modal frequencies and corresponding mode shapes of the internals.

4.3 Acceptance criteria

The acceptance criteria to assess the measurements in comparison with the analytical results are induced from the peak alternating stress considering the maximum alternating stress and stress concentration factor at the maximum stressed location of each component of reactor vessel internals.

5. Conclusions

The integrity of Shin-Kori 3 and 4 RVI will be verified by performing the non-prototype category II CVAP and the limited measurement program in Shin-Kori 4 will be performed differently than the RVI CVAP in OPR1000. To determine measurement locations and acceptance criteria, the dynamic response of the reactor internals against the hydraulic forcing function will be calculated using commercial CFM and FEM codes for the hydraulic and structural analysis.

If the RVI CVAP in Shin-Kori 4 is completed successfully as the non-prototype category II without experiencing any adverse inservice vibration phenomena, Shin-Kori 3 and 4 will be the limited valid prototype for all plants that follows them.

REFERENCES

[1] U.S. Nuclear Regulatory Commission Regulatory Guide 1.20, “Comprehensive Vibration Assessment Program for Reactor Internals during Preoperational and Initial Startup Testing”, Revision 3, March 2007.
[2] Combustion Engineering, INC., “Comprehensive Vibration Assessment Program for the Prototype System 80 Reactor Internals (Palo Verde Nuclear Generating Station UNIT 1)”, 1984.
[3] KEPCO, 10487-ME-TE-240-03, “A Comprehensive Vibration Assessment Program for Yonggwang Nuclear Generating Station UNIT 4”, Rev. 0, August 1995.
[4] KEPRI, TM.99NJ13.P2000.243, “Report on the Observation and Synthesis about CVAP in KNGR RVI”, June 2000.