Flow-induced Vibration of Reactor Internals Structures

Huinam Rhee* and Heuy Gap Song*

Reactor Mechanical Engineering, Korean Atomic Energy Research Institute, Taejon , 305-353, KOREA

A flow-induced vibration assessment program has been performed for a pressurized water reactor in Korea in order to verify the structural integrity of the reactor internal structure for flow-induced vibration during normal and transient operating conditions. Through structural analysis, the theoretical evaluation for the structural integrity of the reactor internal structure and the basis for the necessary measurement and inspection are provided. Flow-induced hydraulic loads and reactor internal structure vibration response data were measured during pre-core hot functional testing. Then, the measured data were reduced to obtain auto- and cross-power spectral densities, coherence and phase information for appropriate sensor signals. After that the reduced data were analyzed and compared with the predicted data obtained from structural analysis. The measured structural frequencies agreed well with the results from the analysis. The measured stresses are less than the predicted values and also less than the allowable limits. It is concluded that the structural analysis procedure used for the reactor internal structure is appropriate and that the vibration response of the reactor internal system due to flow-induced vibration under normal and transient operations is acceptable for long term operation.

* presently with Korean Power Engineering Company at the same address

1. INTRODUCTION

The reactor internal structure (Fig. 1) is the complex structure located inside the reactor vessel of a pressurized-water reactor type nuclear power plant. The reactor internal structure consists of several mechanical substructures such as the Core Support Barrel (CSB), the Lower Support Structure, the Core Shroud, the Upper Guide Structure (UGS) assembly, the Control Element Assembly (CEA), and the shroud assembly, etc. The reactor internal structure has a very important role in supporting and protecting the reactor core during all possible operating conditions. This structure includes the nuclear fuel elements. Significant amounts of static and dynamic hydraulic loads are applied to the reactor internal structure during various transient and normal operating conditions over the 40 years service life expected for the reactor. Therefore, the reactor internal structure should be designed to avoid any excessive vibration due to the flow-induced hydraulic forces caused by the reactor coolant. Recently, a series of vibration-assessment programs[1] has been performed for the reactor internal structure of a pressurized-water reactor type nuclear power plant in Korea during its pre-service period. The purpose of the vibration-assessment program is to verify the structural integrity of the reactor internal structure due to flow-induced vibration prior to commercial operation. This vibration-assessment program consists of several procedures such as structural analysis, measurement of vibration and hydraulic loads, inspection and final evaluation. The flow chart for the overall procedures is shown in Fig. 2. We will briefly discuss each of the procedures in the following sections.

2. STRUCTURAL ANALYSIS

The purpose of the structural analysis is to predict the dynamic responses of the reactor internal structure due to flow-induced hydraulic loads, and also to provide the bases for the measurement and inspection before the actual vibration test. The methodology used to calculate the dynamic responses is divided into three parts: 1) calculation of the hydraulic loads or forcing function, 2) analysis of the structures to determine their modal characteristics (natural frequencies and mode shapes), and finally, 3) calculation of the response such as displacements, strains and stresses. The resulting information on the natural frequencies, stresses, strains and displacements are then compared with the actual measured data.

2.1. Flow-induced Hydraulic Loads

The flow-induced hydraulic loads are classified as either deterministic or random, according to their variation with time. The deterministic and random components are assumed to be uncorrelated. A complete description of the deterministic and random loads includes information on magnitudes, frequencies and spatial distributions. Deterministic loads are caused by the harmonic variations (pump pulsations) in the fluid pressure due to the reactor coolant circulating pumps (RCPs). These pump pulsations propagate throughout the system as acoustic waves. The peaks of the pump pulsations occur at multiples of the pump rotor frequency of 20 Hz and the blade passing frequency of 120 Hz. A combination of mathematical analysis and results from experimental data are used to determine the magnitude, frequencies and distributions of the deterministic loads[2]. We need to solve the acoustic wave equation, which is established by combining the equations of motion, continuity and state. Experimental data are required to determine appropriate boundary conditions needed to solve this acoustic wave equation.

Random loads are generated by flow turbulence. All random loads are assumed to be stationary and ergodic. The power spectral density representations in the form of pressure squared per unit frequency versus frequency are developed. As an example, Figure 3 shows the measured random turbulence power spectral density curve for the Control Element