# The Dynamic Behaviour of Reactor Internals for 900 MWe Reactor in Korea under Seismic Excitation

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## 1. INTRODUCTION

The reactor internals in a pressurized water reactor(PWR) system consists of the complex components such as reactor vessel, core barrel, core barrel flange and fuel assemblies, etc. These components are designed to withstand the accident loads such as loss-of-coolant accident and earthquake. Because of the clearance between various components, i.e the clearance between the reactor vessel and the core barrel and among the fuel assemblies, the impact could occur as a result of their relative motion during accidents. The knowledge of the dynamic behaviour under accident loads is important for the design of the reactor internals components.

For this analysis, in the present study, the 900MWe reactor internals are modeled for seismic analysis as in Figure 1, which is called Global Beam Model, of which the characteristics are obtained by aids of the generalized FEM code 'ANSYS'. The model includes nonlinear impact springs and also hydrodynamical couplings.

After the model is established, the modal analysis is performed to examine whether the inputs are properly prepared and to calculate the structural damping values. The displacement time history converted from acceleration response spectrum of SSE(Safe Shutdown Earthquake) for Young Gwang 1&2 nuclear power plants is applied at reactor vessel support location.

As the result, the magnitude of the maximum impact forces between various components and the stress of core barrel upper flange which is delicate part in reactor internals under external load are obtained and the results are acceptable in the standpoint of design. And the responsive motion of the upper core plate and lower core plate are obtained, which are to be used as an input for the analysis of the detailed fuel assembly series model.

## 2. ANALYTICAL METHOD

The reactor internals model(Global Beam Model) consists of beam elements, the nonlinear impact springs, rotational spring and hydrodynamic couplings : The nonlinear impact springs which simulate the gaps between the lower radial restraint and lower radial clevis insert / between the core barrel upper flange and reactor vessel have nonlinearities because of the gap. The genalized FEM code 'ANSYS' are used to get the spring characteristics of these components. And the rotational spring simulates the torsional phenomenon of core barrel upper part. Since the rotational spring characteristic of core barrel upper part is associated with the stiffness of holddown spring as well as upper support plate flange, the spring characteristic is obtained from the combination of springs of core barrel upper part ,upper support plate flange and holddown spring. The hydrodynamic couplings simulate the fluid-filled gaps between the core barrel and reactor vessel according to the position of cylinder segments. And also the nodes which represent the mass and position of reactor components are linked by beam element. The springs between reactor vessel support and the reactor vessel simulate the support condition of nozzles.

The displacement time history which is used for the excitation motion in this study is converted from the acceleration response spectrum at reactor vessel support[1] by aids of STARDYNE code and integration scheme. The response spectrum and displacement time history are shown in Figures 2 & 3, respectively.

After preparing the input data of the reactor internals model, the verification procedure on inputs is performed. And through this procedure the structural damping could be obtained : The dynamic characteristic analysis of the reactor internals model as shown in Figure 1 is performed. As the results, the fundamental natural frequency is 7.1Hz., and the structural $\alpha$- and $\beta$-damping values which are calculated from the obtained natural frequencies are 1.863E-4 and 1.601, respectively. The governing mode shapes of the model are shown in Figure 4.

For dynamic behavior analysis of the established reactor internals model under the given seismic excitation, the governing differential equations of motion are integrated numerically by way of KWUSTOSS computer code[2] which uses the explicit Runge-Kutta-Gills algorithm in time. This algorithm is especially suitable for the treatment of the impact problems which could be occurred because of the gaps between high frequency interference components such as reactor components.

## 3. RESULTS

The maximum force could be get at the position of core barrel upper part(between node 14 and 16) at 4.14 sec. The calculated stress in this part is 2.97 N/mm². The maximum displacement could be get at the top level of the model(Node 17). Regarding the maximum force and displacement, the core barrel upper flange undergoes negligible stress under seismic excitiation.

Because of the position of the excited node(node 18), the reactor internal model behaves like a pendulum and its dynamic behaviours under seismic excitation are almost same independent on the time. The dynamic behaviour of reactor internals under seismic excitation is formed by the contributions of the second and fourth mode. And the maximum absolute displacement at lower radial clevis insert position where the impact could be occurred is 1.8cm. Comparing this value with the excited input displacement, this amount of displacement will not influence to the structural integrity of the reactor internal components. And the maximum impact force could be get at the position of the gap between core barrel upper flange and reactor vessel(the impact spring connected between nodes 15 and 16) and is shown in Figure 5. The maximum displacements and forces of each node are occured at the vicinity of 4.0sec.,so this value includes the meaning that the movement of reactor internals under seismic excitation keeps away from the resonant phenomenon.

The absolute displacements at upper core plate and lower core plate are obtained as shown in Figure 6, which are to be used as an input for the analysis of the detailed fuel assemblies series model. Considering the structural integraty of fuel assemblies, the relative value between upper core plate and lower core plate is not so large that the movement of plates will not affect.

## 4. CONCLUSIONS

Based on the modelling method of reactor internals for accident analysis, the followings can be concluded :

- The stresses of reactor internal components under seismic excitation is negligiable comparing to the yield strength of the component materials.
- The behaviours of 900MWe reactor internals in Korea during seismic event were subjected to the second and fourth modes in linear modes even if it has nonlinear characteristics.
- The maximum impact on reactor internals occurred at the core barrel upper flange position.
- Even if the displacements of reactor internals under SSE are absolutely large, their relative values are small comparing to the building structure and they move in-phase, which do not result in serious problems on the structural integrity.
- The maximum internal force and impact occurred at the very low frequency band, less than 0.5Hz, in which would not be occurred the resonant phenomenon.

## REFERENCES


1. Korea Electric Power Corporation (1985). Final Safety Analysis Report for Young Gwang Nuclear Units 1 & 2.

2. Stabel and Hübsch (1987). KWUSTOSS code manual.


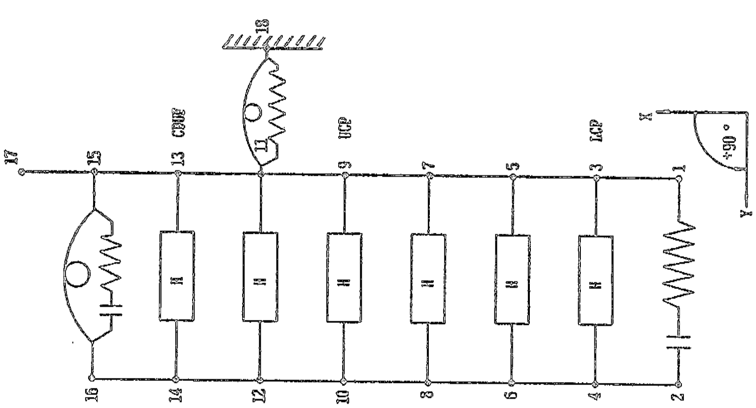


Fig.1 Reactor Internals Horizontal Model

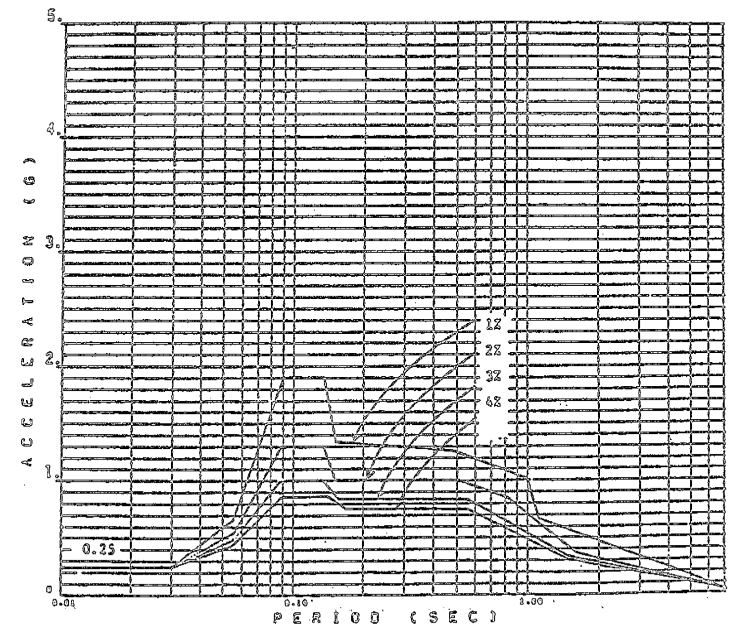

Fig.2 Acceleration Response Spectrum

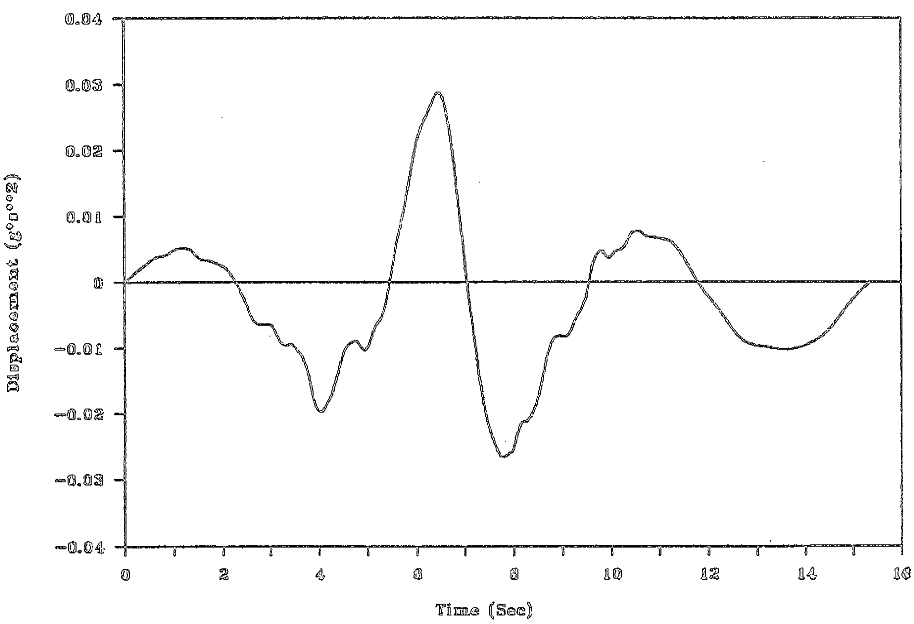

Fig.3 Artificial Seismic Displacement Time History

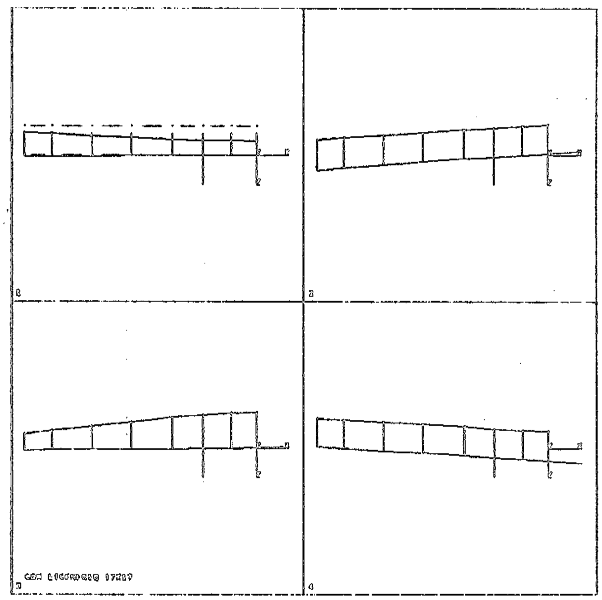

Fig.4 Mode Shapes of Reactor Internals Model

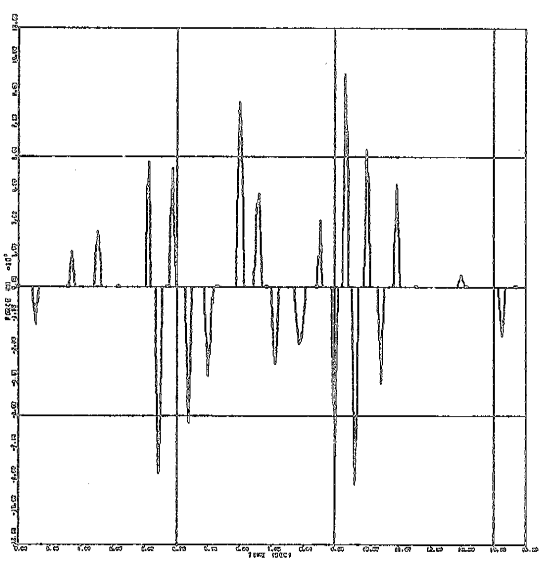

Fig.5 Impact Forces at coupling 15-16

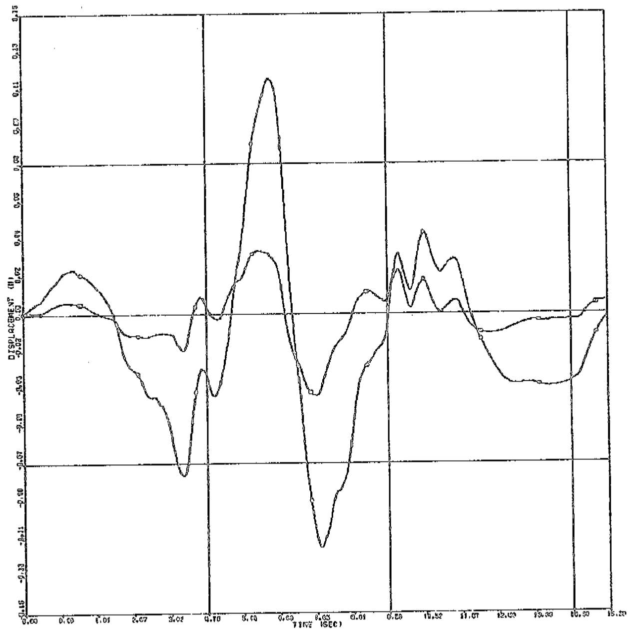

Fig.6 Displacements of nodes 4 & 10